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Working Material

Meeting Report of the

4th 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

Vienna, Austria

17 -21 December 2012

Vienna, Austria; March 2013

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 enhancement 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.

In the above context, a new IAEA CRP on “Innovative Methods in Research Reactor Analysis: Benchmark against Experimental Data on Neutronics and Thermalhydraulic Computational Methods and Tools for Operation and Safety Analysis of Research Reactors” was designed and initiated in October 2008, as a cross-cutting activity jointly operated and equally funded by NSNI, NEFW and NAPC. During the CRP, a great volume of the experimental data was obtained from different CRP participants, covering a wide range of RR types, neutronic and thermalhydraulic parameters, power levels and experimental configurations. This data base is now compiled into the facility specifications, experiment descriptions and corresponding experiment data for 9 research reactors. Each data set is prepared in a way to serve as a stand-alone resource to perform independent benchmark exercises by interested institutions world-wide.

The output of the CRP will be two IAEA publications describing the data, results, analyses and conclusions. Overall, the CRP expected research outcomes were:

- 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.
- Comprehensive database of RR characteristics, experiments and data used for benchmarks
- Recommendations on open issues for future R&D activities involving RRs
- Increased cooperation among RR analysts related to experiments and modelling

According to the CRP Action Matrix, which was developed in the first RCM and updated during the subsequent RCMs, the activities of the Chief Scientific Investigators (CSIs) 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 results of the CSIs modelling efforts have been submitted to the designated results consolidators for further processing. This was done in most of the cases prior to this last RCM.

The main objectives of this 4th RCM were:

- To evaluate the progress made towards achievement of the CRP objectives and outcomes
- Present and discuss consolidated results of benchmarks of nine research reactors
- To agree on the necessary follow-up activities to be held during 2013
- To revise the contents and format of the 2nd draft publication “Research Reactor Benchmarks: Computational Results and Analysis” as an output of the CRP
- To clarify clearance procedures and publication rights for reactor specifications and data publication
- Formulate conclusions and recommendations related to the preliminary analysis of the consolidated results
- To prepare a draft meeting report

B. **Meeting Achievements**

The meeting was attended by 20 participants representing 15 Member States. Mr D. Ridikas, NAPC-PS and Mr A. M. Shokr, NSNI-RRSS were Scientific Secretaries of the meeting. A full list of all participants is provided in Annex 6. The meeting was hosted by the IAEA, in Vienna, Austria and began with 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 5). Mr. A. Olson (ANL, USA) was designated as Chairperson whereas Mr. G. Braoudakis (ANSTO, Australia) and Mr. S. Chatzidakis (INT-RP, Greece) were designated Rapporteurs of the Meeting.

The first three days of the meeting included a brief review by the IAEA representatives of the CRP objectives and the expected results. In addition, the meeting participants were informed of the status of the 1st publication which is under internal clearance process and expected to be published during Q2, 2013. Right after followed by technical presentations by the results consolidators on the status of the work and results achieved so far. Additional individual presentations were provided by France (CEA) and Ghana (GAEC). Summary reports of all individual country achievements are included in Annex 8.

The IAEA representative stressed that data providers should notify IAEA of any restrictions imposed on data or reactor specifications by the end of the meeting. France (CEA) pointed out that publication of data is allowed but experimental fuel pin and fuel composition details should not be shown. As a result after some minor amendments to the data files, all data providers have signed publication clearance (copyright) forms by the end of the meeting. This allowed the Secretariat to go ahead with the 1st publication of the CRP.

The participants discussed the consolidated results and made observations and comments regarding the capabilities of the various codes to provide reliable predictions of the experimental data. During the discussions there were possible explanations on the causes for the observed discrepancies. This assisted the participants to elaborate on follow-up activities necessary to optimize the outcome of the results consolidation for the benefit of the RR community and prepare inputs for the 2nd publication of the CRP. Specific deadlines were set for the final inputs/contributions to be provided and drafting of the 2nd IAEA publication. Other deadlines and action plan are given in Annex 2.

The meeting further discussed the responsibilities regarding collation of results for each benchmark. The third day was concluded with specific breakout sessions during which results consolidators and CSIs could clarify specific issues posed during the technical presentations of benchmark results and their comparison with experimental data. In particular the results consolidators required further details and information regarding specific modelling assumptions and approximations adopted by the analysts. This would assist in understanding some of the discrepancies between calculations and experiment and the various calculations themselves.

The meeting Chairperson commented on the significance of the database which should be widespread to RR community to take advantage and improve safety and operation of research reactor facilities.

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 concluded that the drafting of the 2nd IAEA publication can only be a two-way process. The results consolidators will circulate the first draft to the individual participants to elaborate and review the document. Following acceptance by the individual participants, the results consolidators may finalize the document to be submitted for independent review during a dedicated IAEA Consultancy Meeting (CM) later in 2013.
- 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 7.
- Similarly, the results consolidators were requested to summarize the work status so far, and the conclusions drawn during this meeting per research reactor facility, included in the benchmarking exercise. These conclusions can be found in Annex 7.
- 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.
- In order to ensure effective results consolidation and perform combined analysis of individual results reported by all CRP members, a dedicated CM will be held in Q2-Q3 of 2013. This CM will collate joint technical conclusions particular to the various codes and benchmark problems performed within the CRP, and finalize the 2nd IAEA publication of the CRP.

Further activities during the last two days focussed on the review and finalisation of the results consolidation. The following are highlighted:

- 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 is reached and time allows.
- User effect was identified as a major origin among code-to-code discrepancies. It was concluded that certain input parameters and nodalization methods affect significantly the results provided by the codes. Additional details can be found in Annex 7.
- The coupling between neutronics and thermal-hydraulics via the kinetic parameters and feedback coefficients seems to be the major contributor to discrepancies between

experimental measurements and codes estimates. It is recommended that harmonization effort should be made in this area to improve the results of the models.

- A number of potential ways to reduce user effects were suggested by the participants. Among others, sensitivity analysis and adequate training through code benchmarking were the most prevalent. Additional details are included in Annex 8.

Finally, some time was spent on discussing the possible outcomes and future activities following this CRP. A new CRP to be launched on fuel burn-up and material activation was announced by the IAEA representative. Interested participants should notify the IAEA at the earliest possible.

Meeting participants suggested that the database and the benchmarking effort during the CRP should be made widely known throughout the research reactor community. Continuation of the benchmarking effort should be encouraged and should there exist enough interested participants the IAEA is committed to organize and host a technical meeting to discuss the updated results. In addition, the current database should be a starting point and further facility benchmarks and experiments can be added in the future to increase the value and relevance of the database.

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 results consolidators and 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 all meeting participants to the host organizers and IAEA representatives for their considerable efforts and continuous support provided during the meeting.

C. **Summary of Conclusions**

The activities of the 4th RCM showed 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.

During this meeting, the draft of the second IAEA publication, the table of contents of the second publication, and the follow up activities for participant submissions were reviewed and accepted with minor adjustments. All benchmarks completed and the reports and results have been sent to results consolidators. Results consolidation is still in progress and will be finalized following the submission of the final versions of the individual reports.

Given the scope of this CRP, which includes nine benchmark problems (with at least three experiments each) covering broad subjects related to research reactor safety, operation and utilization, the participants were informed of the 1st publication status. One additional consultancy is planned during 2013 to summarise and collate joint conclusions regarding the results of the benchmarks. Participants have to submit final individual reports by February 15th, 2013.

- The activities reported during the 4th RCM showed that the most important objectives of the CRP were achieved

  - Benchmarks forum is established and includes 20 MSs; this provides the basis to continue the networking
  - 1st RR Data base (1st publication) for benchmarks available and ready for use
  - Benchmark calculations and analysis completed and available as a result of this CRP (to be finalised and reported in the 2nd CRP publication)
  - User effects: evident, identified and discussed as part of the exercise in a number of cases
  - Work needs to be continued, based on recommendations and the agreed work-plan

- The status of facility and experiment descriptions (benchmark specifications) were further improved and clarifications were gathered prior to final 1st IAEA publication containing this information. WWR-M (Uzbekistan) specifications have been excluded from the benchmarks as not fulfilling the requirements. This change resulted in nine remaining benchmark problems (with at least three experiments each) covering broad subjects related to research reactor safety, operation and utilization. Copy right agreements have been collected from all data providers, allowing the IAEA to proceed with the 1st publication immediately.
- The final draft of the 1st IAEA publication, the table of contents of the second publication, and the results templates for individual participant submissions were confirmed. All benchmarks have the required minimum contributions (more than two). SPERT-IV benchmarks attracted the most numerous number of interested participants (6).
- The participants have set a clear action plan to finalise the individual reports, consolidated reports and prepare the 2nd publication “Research Reactor Benchmarks: Computational Results and Analysis” by July 2013.
- Follow up actions have also been discussed, including organization of related technical meetings/workshops and a start of the new CRP on “Innovative methods in Research Reactor Analysis: Benchmarks against Experimental Data on Fuel Burn-up and Material Activation” (2014-2017)

In addition, the following conclusions were drawn by participants:

- The CRP is helpful to the research reactor community **thanks to the CRP participants** for making this effort possible. **The CRP has provided a unique 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 performing preliminary analysis on all of these, it was noted that **interactions 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**

- 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 to continue the benchmarking process. Interested participants are welcomed to share their updated results in a dedicated meeting. IAEA is committed to organize and host such a technical meeting following participants' request.

- The definition of quality of the benchmarks is a difficult task, and often shortcomings are only found 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.**

- **Good communication between data suppliers, analysts, and code developers should be continued and facilitated**. For example, some input decks will be shared among the analysts and the code developers for cross check and advice as necessary.

- **The following briefly resumes individual RR facilities benchmarked:**

  - **OPAL:** no major issues in this benchmark; some variation in the results obtained for <u>control rod worth and kinetic parameters;</u> specifications are of good quality and can serve for other teams as a good source for code benchmarking/qualification;

  - **MNR:** issues in reproducing point kinetic parameters; some of the benchmarks specifications are not well defined and not self-consistent; uncertainty in fuel burn-up specifications is important; <u>fresh versus used fuel core</u>; support for future CRP on burn-up; <u>identify drawbacks/pending issues for each benchmark in report 2</u>

  - **SPERT-IV D-Core:** statics data/experiments are well qualified; transient benchmarks are the most challenging; 6 teams participated; large deviations and discrepancies have been observed between experiment and calculations and between the various calculations; <u>benchmarks and analysis should cover both short and long period ranges of transients; additional SPERT-I A, B and D-cores are available for further benchmarks.</u>

  - **ETRR-2, RSG-GAS:** overall data specifications are of acceptable quality; generally, keeping in mind conservatism, the results were of good quality; <u>user effects were observed by comparing results produced by the same codes</u>; <u>in other cases, trends in the calculation results suggested limitations of some modelling tools;</u> sharing of inputs decks was recommended to assist in identifying the cause of the observed variation, and perhaps initiate/support training; importance of sensitivity analysis for some parameters was emphasised; <u>refining of calculations and comparison is</u>

indispensable in some cases; in practice, instrumented fuel elements should be prioritized when possible/justified; in addition to conservative approach, importance of comparing the evolution of transient parameters; issue with the predictive power (thermal-hydraulic) for unknown solutions.

- **MINERVE**: overall definition and availability of uncertainties regarding specifications are of high importance to be assistance in interpreting results. Predictions were of good quality; need for verification of dependence of results on nuclear data libraries (e.g. Mo-capture).
- **MNSR-Y** & **IEA-R**1:
- code modifications are required to include improved correlations and 3D effects; instrumented fuel elements recommended; recommended to continue the efforts beyond this CRP; conservatism was observed in most of the calculation results; some uncertainties in temperature measurements close to the fuel cladding; more consolidated work to come.
- user effect was observed using RELAP; share/comparison of input decks should be the first action to resolve this discrepancy.
- **Observation on RELAP code:** should not be used in 1-D mode for situations where 3D phenomena (e.g. natural circulation) are important. Such cases, being identified, should be followed up in order to formulate more definitive conclusions and recommendations.

D. Summary of Recommendations

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

1. D**efine 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 evaluate the calculational capabilities of modern code systems, in particular against recent research reactor designs.
2. C**onsider the development of training material or a course based on the data and models developed and used during the CRP; promote the database to interested parties outside the CRP as soon as it is finalised and available.**
3. Create and maintain a dedicated web-portal for follow up activities to support the present CRP community as well as other interested parties in terms of knowledge management, sharing of experiences and good practices in benchmarking of RRs as well as use of various modelling tools.
4. A**llow future submissions to the IAEA database in regard to both additional benchmarks as well as further calculated results on existing benchmarks**, and hence consider revised publications in the future.

5. Provide support and **organize a follow up technical meeting/workshop** on progress and new developments in benchmarking of the RR experiments, collected and made available during this CRP.

6. O**rganize code or benchmark specific user group meetings or facilitate other means of communication among the participants** to enhance the quality and outcome of the final submissions to the second IAEA publication.

7. Propose, define and initiate **a research reactor based depletion benchmark CRP**, in particular to address issues related to efficient reactor fuel utilization, source term definition, waste quantification, irradiation of various targets and other topics of interest to safety and utilisation of RR.

ANNEX 1: 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, SAF, KOR	ARG, AUL, SAF, KOR	ARG, AUL, SAF, KOR	ARG, AUL, SAF, KOR		ARG, AUL, SAF	ARG, AUL, SAF	ARG, AUL, SAF				
ETRR-2	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF	ARG, SAF			ARG, GRE, EGY, SYR, SAF	ARG, GRE, EGY, SYR, SAF	ARG, GRE, EGY, SYR, SAF
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										ARG, GRE, EGY, SYR	ARG, GRE, EGY, SYR	
SPERT-III	GHA, ROM, USA	GHA, ROM, USA	GHA, ROM, USA	GHA, ROM, USA	GHA, ROM, USA	GHA, ROM, USA	GHA, ROM, USA					GHA, 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, PAK, SYR

IEA-R1										**ALG, ARG, BRA, GRE, Korea, SYR, BGD**	**ALG, ARG, BRA, GRE, Korea, SYR, 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),
Greece (GRE) Pakistan (PAK), Romania (ROM), South Africa (SAF),
Syria (SYR), United States of America (USA).

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

Note: Italy, Nigeria and Uzbekistan have initially participated and contributed to the CRP but left the project at the later stage.

ANNEX 2: LIST OF THE FOLLOW-UP ACTIONS

Deadline	Action	Responsible
15/01/2013	Final input regarding facility and exp. spec. (report 1); after this date web-page will be released for public use as working document; the official IAEA publication will follow right after.	Data providers
15/02/2013	Final individual contributions/results sent to the IAEA & results consolidators; the authors should prepare their individual reports strictly according to the agreed contents and format (see templates)	All
15/04/2013	Consolidators send their consolidated reports to individual contributors for review; the authors should prepare their reports strictly according to the agreed contents and format (see templates)	Result consolidation team
15/05/2013	Deadline for accepting reviews/suggestions on consolidated report from all	All
June-July 2013	Final review meeting (publication 2) in Vienna	IAEA
Sept.-Oct. 2013:	After internal clearances, the publication is submitted to the publication committee as Nuclear Safety Series document	IAEA

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
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

ANNEX 3: TABLE OF CONTENTS FOR THE CRP PUBLICATION PART II (BENCHMARKING RESULTS)

Deadline: 15 April 2013

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 4: TABLE OF CONTENTS FOR THE FINAL REPORT OF THE INDIVIDUAL COUNTRY REPORTS

Deadline: 15 February 2013

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 5: MEETING AGENDA

Tentative Agenda

4th 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

17 - 21 December 2012, IAEA, Vienna, Austria

Meeting Room M5

Monday, 17 December 2012		
09:30-10:00	- Welcome by the IAEA - Self introduction of the participants - Adoption of the Meeting Agenda - Election of Chairperson/Rapporteur	IAEA ALL ALL ALL
10:00-10:30	- Status of the CRP - Status of the 1st publication - Already planned follow up activities - Expected output of the 4th RCM	IAEA
10:30-11:00	Coffee break	
	Joint Presentation by the Chief Scientific Investigators (CSIs), responsible for consolidation of the results for:	
11:00-12:30	1. RSG-GAS, followed by discussion	GRE
12:30-14:00	Lunch Break	
14:00-15:30	2. IEA-R1, followed by discussion	BRA, SYR
15:30-16:00	Coffee Break	
16:00-17:30	3. SPERT-IV, followed by discussion	CAN, SAF, GRE

Tuesday, 18 December 2012		
	Joint Presentation by the Chief Scientific Investigators (CSIs), responsible for consolidation of the results for:	
09:00-10:30	4. MINERVE, followed by discussion	FRA
10:30-11:00	Coffee break	
11:00-12:30	5. SPERT-III, followed by discussion	USA
12:30-14:00	Lunch Break	
14:00-15:30	6. OPAL, followed by discussion	AUS, ARG

15:30-16:00	Coffee Break	
16:00-17:30	7. ETRR-2, followed by discussion	EGY, ARG, GRE
19:00	Hospitality event	ALL

Wednesday, 19 December 2012		
	Joint Presentation by the Chief Scientific Investigators (CSIs), responsible for consolidation of the results for:	
09:00-10:30	8. MNR, followed by discussion	CAN
10:30-11:00	Coffee break	
11:00-12:30	9. MNSR-Y, followed by discussion	SYR, USA
12:30-14:00	Lunch Break	
14:00-15:30	Final concluding presentations by Chief Scientific investigators (CSIs), responsible for consolidation of the results: 30min per research reactor benchmarked	RSG-GAS, IEA-R1, SPERT-IV
15:30-16:00	Coffee Break	
16:00-17:30	Final concluding presentations by Chief Scientific Investigators (CSIs), responsible for consolidation of the results: 30min per research reactor benchmarked	MNSR-Y, MINERVE, SPERT-III
Thursday, 20 December 2012		
09:00-10:30	Final concluding presentations by Chief Scientific Investigators (CSIs), responsible for consolidation of the results: 30min per research reactor benchmarked	MNR, OPAL, ETRR-2
10:30-11:00	Coffee break	
11:00:12:30	Consolidation of the status and progress made in relation to the achievement of the CRP objectives	All
12:30-14:00	Lunch Break	
14:00-15:30	Discussion/formulation of summary and conclusions for the whole CRP	
15:30-16:00	Coffee Break	
16:00-17:30	Discussion/formulation of summary and conclusions for the whole CRP	All

Friday, 21 December 2012		
09:00-10:30	Discussion and drafting the RCM report, including recommendations and follow up activities	All
10:30-11:00	Coffee break	
11:00:12:30	Finalization of the RCM meeting report	All
12:30	Closing of the Meeting	

ANNEX 6: LIST OF PARTICIPANTS

4th 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

17 - 21 December 2012, IAEA, Vienna, Austria

Meeting Room M5

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ANNEX 7: SUMMARIES OF RESULTS CONSOLIDATION

1.1 SPERT-IV DATA CONSOLIDATION REPORT – STATIC SECTION (Canada)

The participation in the SPERT-IV Benchmark Problem Static section is as follows:

Group	Code	ICE & OL	Rod Worth	Void Reactivity	Temp Reactivity	Flux*	Kinetics Params
AUS	MCNP5	Yes	Yes	Yes	Yes	Yes	Yes
BGD	MVP	Yes	Yes	Yes	-	Yes	Yes
FRA1	TRIPOLI-4	Yes	Yes	Yes	Yes	Yes	Yes
FRA2	SCALE 6.0	Partial	Yes	Partial	-	-	Yes
PAK	MTR-PC	-	Partial	Partial	Partial	Partial	Yes
SYR	MCNP-4C	Yes	Yes	Partial	Yes	Partial	Yes

Submitted results show reasonable agreement with experimental results. Some variation was found in the kinetics parameters calculated/adopted by the different groups. This affects the conversion of calculated results into the units of the experimental data and would similarly affect the input data to the companion transient calculations. Initial critical experiment, critical rod positions, excess reactivity calculations, and control rod worth calculations were performed successfully with minor user effect noted. Temperature coefficient results highlighted the difficulty in calculating small changes in reactivity with Monte Carlo codes due to the small magnitude of the effect compared to the statistical uncertainty of the calculation. This is common to other benchmark problems. Comparisons of flux distributions were not completed at this time and require participants to resubmit results with a standardized normalization and averaging.

Requests were made for further information on modelling details required to complete a comprehensive data consolidation/assessment. In addition, sensitivity analysis results were requested of the AUS and FRA1 groups as this information was not retained in their final country reports. Outstanding reports and data spreadsheets were requested of FRA2, PAK and SYR.

In general it can be concluded that the Static section of the SPERT-IV Benchmark Problem represents a good range of routine neutronics calculations and provides a useful and comprehensive analysis problem for comparison to experimental results. Limitations in the Benchmark Problem do exist but are minor in extent and significance and can be identified as areas for sensitivity analysis.

1.2 SPERT-IV CONSOLIDATION REPORT – TRANSIENT SECTION (Greece)

The participation in the SPERT-IV Benchmark Problem Transient section is as follows:

Table: Codes used by participating groups

Group	Transient (RIA) Codes
Australia (AUS)	PARET-ANL
Bangladesh (BGD)	EUREKA-2/RR
Greece (GRE)	PARET-ANL

France-CEA (FRA)	CATHARE-2
France-IRSN (FRA2)	ASTEC
Syria (SYR)	MERSAT/RELAP

Preliminary conclusions:

- There are a lot of subtleties involved in developing an input deck and the problem should not be considered purely thermal-hydraulics but rather a coupled neutronics/thermal-hydraulics problem.
- Comparison among different participants, different codes and different modeling philosophies.
- Some participants performed blind calculations, some tuning, some were purposely conservative for safety analysis.
- The results represent the modeling state of analysis capability in RR community (it is not possible to say which is the best model or the best code).
- In some cases the qualitative behaviour of the transient was captured in the simulation results but this was limited.
- Different nodalizations (hot channel and lumped average channel vs. multichannel model).
- Different methodologies followed (built in models vs. custom models, including PCS or not).
- Different input parameters (misinterpretation of specifications, assumptions, unknown values).
- Some parameters more important than others (such as feedback coefficients).
- User effect was identified that is transient-dependent.
- Sensitivity analysis and adequate training are recommended as possible ways to reduce user effect. Emphasis on this being a coupled neutronics/thermalhydraulics problem was identified as opposed to a purely thermalhydraulics problem.

2 MNR DATA CONSOLIDATION REPORT (Canada)

Three groups participated in the MNR Benchmark Problem as part of the IAEA CRP 1496: ARGENTINA, CANADA, and SOUTH AFRICA. Extent of participation is summarized below:

Group	RR Calib	SR Calib	Misc ρ Tests	Radial Wires	Axial Wires	Void Expt	Pool Temp
ARG	Yes	Yes	-	-	Yes	Yes	Yes
CAN	Yes	Yes	Yes	Yes	Yes	-	-
SAF	Yes	Yes	-	Yes	Yes	-	-

The Argentina (ARG) and South African (SAF) groups have successfully calculated the sections of the problem in which they participated. There were some problems with the criticality calculations performed by Canada (CAN) so these results will be reviewed and, if time permits, corrected. If this is not possible some of the CAN results will be withdrawn from the final publication.

With respect to the different sections of the problem: There were some notable differences in the conversion of doubling times into reactivity values related to the choice of point kinetics parameters used by the groups. This remains unexplained at this time and warrants further investigation.   Minor effects of modelling approach were noted in the analysis of the axial flux wire section of the problem but overall, the finite difference and nodal diffusion methods provided equivalent results to the more detailed Monte Carlo full-core transport solution. As well, for Cu-wire activation, estimated thermal-group flux solutions were found to be equivalent to flux with detector constant solutions as well as the explicit comparison of activation estimates.   Following further consideration it was concluded that the quality of the MNR Benchmark Problem may be low in the Radial Wire section of the problem. These operational measurements are associated with large uncertainty.

The MNR Benchmark Problem represents a set of rather routine neutronics calculations. Similar exercises are available at higher quality via the OPAL Benchmark Problem. A notable difference in the MNR and OPAL Benchmark Problems is the difference of an operated *vs.* fresh core. However, given the uncertainties associated with MNR burn-up estimates this difference as presented by this specification has limited value. It is noted and supported that a future CRP initiative involves comparison to burn-up data.

The MNR Benchmark Specification is somewhat compromised by the quality of the data in some sections and could be improved by modification of the experimental/measurement procedure to limit uncertainties and provide more reference comparison points with the measurements. Some recommendations for future design of a replacement set of experimental data (from MNR or equivalent) are as follows:

1. Add a reference point to establish point kinetics parameters for an exposed/equilibrium RR core
2. Shim-safety rod calibration via period method rather than cross calibration
3. Second method for rod worth measurement
4. More controlled radial wire map experiment

The exercise of calculating the problem and consolidating data has provided improved insight into the design of such a benchmark problem in terms of points of comparison to experimental data and associated calculation methodologies. The MNR Benchmark Specification and associated results template will be adjusted to capture some improvements/clarifications. It was also decided to remove the supplied fuel material composition estimates from the available problem distribution as it adds little value. Users will rely on supplied depletion estimates per fuel assembly and will be required to calculate associated fuel material compositions for their models.

3 RSG-GAS CONSOLIDATION REPORT (Greece)

The RSG-GAS Benchmark Problem is divided into two sections: (i) steady state measurements, and (ii) transient measurements (Loss of flow). The participation for this problem and the tools used are summarized in the following table.

Table: Codes used by participating groups

Group	Steady State	Transient
Argentina (ARG)	RELAP5/MOD3.2	RELAP5/MOD3.2
Egypt (EGY)	RELAP5/MOD3.4	RELAP5/MOD3.4
Greece (GRE)	PARET-ANL RELAP5/MOD3.3	PARET-ANL RELAP5/MOD3.3
Syria (SYR)	MERSAT RELAP5/MOD3.3	MERSAT RELAP5/MOD3.3

A. Identification of main cause of discrepancies:

(i) Against measurements

- Not clear specifications of thermocouples position. Thermocouples position was agreed among participants during the previous CRP (France, 2011)
- New specification of thermocouples position (Vienna, 2012)

(ii) Among participants

- Different nodalizations (hot channel and lumped average channel vs. multichannel model)
- Different methodologies followed (built in models vs. custom models, including PCS or not)
- Different input parameters (misinterpretation of specifications, assumptions, unknown values)
- Parameters to be paid attention:
    - fuel and cladding thermal properties
    - trip and actuation delay times
    - actuation time (for flapper valves)
    - hydraulic parameters (important for natural convection)
        - geometry
        - friction factors
        - water height above and below the core
- Potential not important (may lead to confusion)
    - kinetic parameters
    - feedback coefficients

B. Overall agreement

- Trend well followed
- Peak temperatures are conservative and occur at the correct time (MERSAT code had a time shift – to be checked)
- Natural convection was consistently overestimated (PARET underestimated – to be checked)
- Decay heat table ANS79 from fission of uranium is acceptable and can be used reliably

C. Preliminary conclusions

- TH codes can be used reliably to simulate mild LOFT if:
- In core channel modeling (1-D) are well simulated
- Out of the core modeling where 3D natural circulation phenomena take place, the results should be used with caution.
- User effect is apparent but not limiting
- No code limitations

D. Neutronics vs TH

- Both SS cases had very good accuracy
- TH steady state can be reproduced accurately
- TH depend on correlations, time-dependent, lumped parameters, averaged calculations, 3D effects

E. Preliminary Suggestions

- sensitivity analysis (for flapper valves)
- adequate training
- underspecified or overspecified specifications to be avoided
- User effect transient dependent
- Nodalization seems to be important

4 ETRR-2 CONSOLIDATION REPORT (Greece)

The participation for the ETRR-2 benchmark problem and the tools used are summarized in the following table.

Table: Codes used by participating groups

Group	Steady State	Transient
Argentina (ARG)	RELAP5/MOD3.2	RELAP5/MOD3.2
Egypt (EGY)	RELAP5/MOD3.4	RELAP5/MOD3.4
Greece (GRE)	RELAP5/MOD3.3	RELAP5/MOD3.3
South Africa (SAF)	RELAP5/MOD3.3	RELAP5/MOD3.3
Syria (SYR)	MERSAT RELAP5/MOD3.3	MERSAT RELAP5/MOD3.3

A. Identification of main cause of discrepancies:

- The temperature inside the chimney is largely dependent on radial position.
  - radial location of thermocouple
  - radial power density
  - natural convection effects not represented
- It is observed that for temperatures outside the core, participants perform crude nodalization. Only after comparison with the results it was discovered that a more detailed nodalization of the outlet plenum and the chimney provides better results.
- Differences in flapper valve and piping models is responsible for discrepancies among RELAP5 users

5 OPAL CONSOLIDATION REPORT (AUSTRALIA)

As consolidators for the OPAL results, reports were requested and received from four groups with a total of six different sets of results (Necsa provided three sets of results using different codes). These results were combined into a consolidated report and analyses performed to evaluate the capabilities of the various codes, identify limitations in the numerical methods adopted and any shortcomings or possible anomalies in the reactor specification or experimental results provided for the benchmark.

The results provided for the OPAL benchmark were obtained using codes based on a range of different numerical methods including the nodal, diffusion and Monte Carlo methods. This allowed a comparison of both different codes and numerical methods. The following comments and recommendations are made regarding the consolidation of the OPAL results.

Spatial flux distribution: All of the codes were able to reproduce the general shapes and characteristics of the various distributions. The thermal neutron flux depends on the local geometry and conditions and these effects are best represented by the Monte Carlo codes. Indeed these codes yield the best results for this experiment. The diffusion models provided the next best results and finally the nodal method. This relative behaviour of the various numerical methods is expected. The significant uncertainty in the reactor power during the measurement led to the various groups adopting slightly different methods to normalise the flux results. This introduced an added slight inconsistency in comparing the calculated results. In addition the calculated results revealed possible errors or inconsistencies in some of the data points related to the upper or lower extremities of the profiles. This should be investigated further by the data provider.

Critical configurations: The calculated reactivity of the 74 critical configurations for five sets of calculations was submitted. All codes provided reliable results with the Monte Carlo method providing the best average but the diffusion code the least standard deviation (most consistent results). It is interesting that modeling differences and slight differences in the code version can produce a difference of 100-150 pcm in the diffusion results. The core reactivity is a global parameter that depends on many variables but seems to be insensitive to finer details. All the codes have demonstrated the capability to estimate core reactivity well within 1000 pcm. The set of data appears to be a valuable addition to other criticality benchmarks.

Control rod worth: The compact core of the OPAL reactor along with the presence of burnable poison in the form of cadmium wires presents a challenge for the calculation of control rod worth as these cause significant fluctuations in flux over small changes in distance. These calculations are also very sensitive to the representation of the geometric and material details of the control rods and surrounding regions. Depending on the calculation scheme employed and approximations adopted the accuracy of the results can vary. The diffusion codes provided the best results with the Monte Carlo method the next most reliable. The nodal method and coarse particle Monte Carlo method provided the most inconsistent results. In addition, there was a difference in β_{eff} between the diffusion code and the others. The origin of this difference has not been resolved but details of the delayed neutron constants adopted by all codes is being investigated.

Fuel burnup: Two sets of complete results and one partial set were submitted for the evaluation of OPAL operational data. The complete sets were based on the diffusion method. They were able to track the operation of the reactor through the six cycles provided with improved agreement tracking the actual reactor power as a function of time. There is a trend in the calculated reactivity with operation time that appears to improve the agreement. The nodal method was also able to track the operational data.

Kinetic parameters: Three sets of results for the prompt neutron decay constant were submitted. Agreement amongst the calculated values is good and also with the measured value. As noted previously the calculated values for β_{eff} indicate significant discrepancies that are yet to be resolved.

Outstanding work

During the presentations and discussions of the OPAL results is became apparent that the specification for the Cold Neutron Source during some of the experiments was not correct. This will be corrected in the next revision of the Reactor and Experiment Specifications.

The discrepancy in β_{eff} is to be investigated by the various contributors by clarifying the origin of the delayed neutron data.

General comments

The OPAL reactor specification and experiments are a valuable addition to research reactor benchmarks.

The Monte Carlo method was able to provide results that agreed very well with measurements as the finer geometry details were well represented for the compact OPAL core. The diffusion method was also able to provide reliable results for the benchmarked parameters.

There was a small user effect evident but this did not produce any significant differences for the benchmarked parameters.

6 MNSR-Y
Not available

7 SPERT-III
Not available

8 MINERVE CONSOLIDATION REPORT (FRANCE-CEA)

The VALMONT program was analysed by Romania (INR) and by France (CEA and IRSN). CEA contributed as provider of the experimental data and also with a new interpretation. Experimental data concern physics measurement (spectrum index and conversion ratio) and safety measurements (control rod worth, radial and axial fission rates, and experimental fuel pin worth).

All the participants chose the Monte-Carlo route for their works:

- INR(Romania): Code MCNPX with ENDFB7 nuclear data library
- CEA (France): Code TRIPOLI4 with JEFF3.1.1 nuclear data library
- IRSN (France): Code MORET with ENDFB7 nuclear data library

IRSN work is a very preliminary analysis and data appear to be inconsistent. The final report was not provided. IRSN hopes to be capable to provide new results and its individual report for the deadline (February 2013)

- INR the interpretation is almost complete, some data are to be verified and additional/corrected data should be provided before the deadline
- CEA made a new interpretation of its program using JEFF3.1.1 as nuclear data library in addition to its previous work with JEF2.2

Safety measurements

Valmont pin weight

- IRSN: there is an identified overlook in the modelling of the experimental fuel
- INR: the discrepancy of -10.72% from the experimental is due to an error in the diameter of the central hole of the experimental fuel and will be corrected.
- CEA: the discrepancy of 5.3% is within 1σ (corresponding to the quadratic sum of standard deviations on the Tripoli4 calculations and on the experimental results).

Radial fission rates distribution

- IRSN: the important discrepancy from the experimental value is consistent with the results on the pin worth and due to the same error in the modelling
- INR: the discrepancy of 5.8% is consistent with the one on the pin worth and also due to the error on the diameter of the central hole of the fuel pin.
- CEA: the discrepancy is consistent with the experimental uncertainties and the standard deviation of the Tripoli4 calculations.

Axial fission rates distribution

- IRSN: No results provided
- INR and CEA: results are consistent with the experimental uncertainties and the standard deviation of the Tripoli4 calculation.

Physics measurements

Spectral indexes and Conversion ratio

- IRSN: due to the important standard deviation of the calculations it is not possible to analyse the results on spectrum indexes. Concerning the conversion ratio, the discrepancy from the experimental values are not consistent between homogenous and Valmont configurations.
- INR and CEA: results are consistent with the experimental uncertainties and the standard deviation of the Tripoli4 calculation.

Conclusions on consolidation on MINERVE experiments

- As a general recommendation, Monte-Carlo calculations for this type of measurements need to have a standard deviation lower than 1% for reaction rates and lower than 5 pcm for reactivity effect (example: for Valmont pin worth).
- The difference of codes seems to have a low impact on the results. The nuclear data library could have an impact that is impossible to evaluate without a good converge of the calculations. Based on this recommendation, a sensitivity study on the impact of the nuclear data would be interesting.

General conclusions

Whatever the field of study (neutronics or thermal hydraulics) supply of experimental uncertainties is fundamental. In addition uncertainties on the calculated value have also to be known, those two requirements allow a correct analysis of the discrepancies of calculated values from experimental value.

9 IEA-R1

Not available

ANNEX 8: SUMMARIES OF WORK STATUS FROM PARTICIPANTS

Australia

The specifications of the OPAL reactor and experiments were revised following comments and feedback from the consultancy meeting and were finalised as part of the first publication. It is anticipated that some minor revisions will be made in response to comments made during the 4RCM to clarify the state of the CNS during some of the experiments.

All calculation results related to OPAL experiments have been completed and submitted for consolidation. There were no major issues arising from these results in terms of direct comparisons with the experimental data available. Some minor issues were raised and these are discussed in the comments regarding the consolidation of all results.

The SPERT-IV neutronic calculations were refined following discussions at the 3RCM and these were finalised and submitted for consolidation. SPERT-IV transient analyses were initially performed using RELAP but these were not submitted for evaluation. It was decided based on results from other participants using RELAP and following the standard methodology for RIA analysis that the SPERT-IV transient experiments will be reanalysed using PARET. The latest version of PARET (v7.5) was requested and obtained from ANL. A total of 31 of the 41 transients were analysed for the study. The remaining cases did not yield reliable results and were not included in the report. It is noted that the requested details regarding location of voiding strips within the coolant channels for the uniform void coefficient are not available and the users must adopt their own strategy to deal with this issue.

Preliminary results for the ETRR-2 steady state thermal hydraulic benchmark were presented and reported at the 3RCM with a commitment to analyse the transient cases. Following presentations and discussions at the 3RCM is appeared that there was some uncertainty in some of the experimental specifications and it was decided not to proceed further with this benchmark. In addition, the preliminary results for the steady-state cases will not be formally submitted for inclusion in the final publication.

Brazil

Brazil provided experimental results performed with the Instrumented Fuel Assembly that was assembled with the following goals:

- to perform more accurate safety analysis for the IEA-R1 reactor;
- to measure the actual cooling conditions, mainly in the outermost fuel plate;
- to validate computer codes being used for thermal hydraulic and safety analysis of research reactors.

Two series of experiments were performed:

1) The reactor power was stabilized at 3.5 MW, 4.0 MW, 4.5 MW and 5.0 MW for some minutes; after the reactor power was reduced in steps and operated for a few minutes in the following powers: 3.5 MW, 3.0 MW, 2.0 MW and 1.0 MW and finally the reactor power was again increased to 3.5 MW, and after stabilization at this power the pumps were turned off.

2) This series of experiments may be divided into two steps:

a) First: with a box around the core, with the objective to eliminate the cross flow between fuel assemblies, and

b) Second: without a box around the core.

These results were simulated by Brazil, Syria, Korea, Greece, Algeria and Bangladesh.
With the results of the IFA we can say that we have answered the main questions. The main conclusions of Brazilian team are:

- Proved that the flow rate of cooling between two fuel elements is actually smaller and eventually may compromise the cooling of the same, although the temperature values recorded are not exactly the value of the surface temperature of the fuel plate.

- The IFA is located in a position with low neutron flux. We observe that lateral plates are more heated, so these differences may be much higher for the hottest channel

- The clad temperature recorded represents approximately the average temperature between surface temperature and fluid temperature. These results may explain the large differences in the results obtained by some teams.
- The installation of a box around the core of IEA-R1 and in others of the same type is a very good solution to reduce the temperature of the external plates.

New experiments will be performed with other pump flow rates and reactor powers.

Greece

By the end of the CRP, Greece has completed all of its commitments. Among them the following actions were performed during the last year:
- Submission of RSG-GAS results template
- Submission of ETRR-2 results template
- Submission of SPERT-IV results template
- Submission of SPERT-IV Final Progress Report in collaboration with Canada
- Submission of RSG-GAS Final Progress Report
- Submission of ETRR-2 Final Progress Report
- Submission of IEA-R1 Final Progress Report

Greece has also undertaken responsibility for results consolidation of three research reactor benchmarks, namely: RSG-GAS, ETRR-2 and SPERT-IV. The preliminary conclusions of the results consolidation procedure were presented during the meeting and discussed with the meeting participants. The conclusions are:

- TH codes can be used reliably to simulate mild LOFT when in-core channel modeling (1-D) is modeled.

- In case of out of the core modeling where 3D natural circulation phenomena take place, results should be used with caution.

- User effect is apparent but not important in mild transients

- Sensitivity analysis, especially when it comes to flapper valves, has been proven to increase accuracy

- Adequate training is necessary to improve analyst capabilities and reduce user effect

- Underspecified or overspecified specifications should be avoided as they can be sources of confusion

Greece is committed to contribute towards the drafting of the 2nd IAEA publication through the results consolidation process. The analysis of consolidated results will be described in a dedicated consolidation report summarizing the main methods, codes and results obtained among different participants throughout the CRP.

France-CEA

The neutronics statics measurements

The whole experimental configurations have been analysed using the CEA's Monte-Carlo simulation code TRIPOLI4 jointly with the nuclear data library based on JEFF3.1.1 evaluation.

All the static neutronic measurements have been interpreted except for the convention ratio for which it had not been possible to find an accurate modelling allowing the convergence of the flux calculation using Monte-Carlo simulations; all the neutronics parameters have been calculated.

The reactivity changes during the fuel loading are calculated. The calculated results are consistent with the measured ones. The critical mass for 21 fuel assemblies and the excess of reactivity corresponding to the loading of 4 more assemblies are also consistent with the benchmark.

The kinetics parameters $\beta_{effective}$ and l* were calculated with a function recently implemented in TRIPOLI4.

The calculated integral and differential rod worth are comparable to the experimental values. The differential rod worth varies almost linearly and decreases with the insertion of the rods. The integral rod worth is calculated to be 5.1$ and should be compared with the experimental value equal to 5.3$. The discrepancy is lower than 4%.

With regard to axial fluxes distributions, the calculations overestimate the flux in the vicinity of the end boxes of the assemblies. This overestimation is due to the lack of information on the exact geometry of this component. In the central vertical zone of the core the agreement between the simulations and the measurements is good.

The isothermal temperature coefficient is very low in the range of the experiment. As the reactivity effect is reduced (few pcm) a Monte-Carlo code such as TRIPOLI4 is not appropriate because of the importance of the standard deviation compared to the reactivity effect itself. As a consequence the values calculated with TRIPOLI4 have an important uncertainty in a range of 7% to 27%.

The void coefficient is very well calculated with TRIPOLI4 at -0.0806 cents/cm3 and the experimental value is given at 0.0807 cents/cm3. Moreover the radial calculated void importance is consistent with the benchmark. The discrepancy is lower than 2% for the peripheral assemblies and about 4% for the central locations. This point is very important because occurrence of void in the core is an important feedback effect that stops the power increase during the reactivity transient.

This work shows the capability of TRIPOLI4 with its associated nuclear data library based on the European evaluation JEFF3.1.1 to calculate different neutronics parameters with a great accuracy.

Thermal-hydraulics transients

The thermal-hydraulics analysis of the SPERT-IV tests was performed using the CATHARE2 code. CATHARE2 is a two-phase flow, one dimensional code used in France for the safety analysis of Pressurized Water Reactors (PWR).The objective was to analyse the tests so as to assess the code in the field of Reactivity Insertion Accidents (RIA). The first analysis performed with the standard version of the code indicated large discrepancies between the calculation and the data. It is as though the heat exchange and the critical heat flux correlations are not adapted to the fast temperature transients as observed in the SPERT-IV tests.

A specific methodology was developed for correcting the available correlations implemented in the code. This methodology was based on iterative calculations using CATHARE2 with imposed nuclear power as measured in the tests. Corrective factors were defined to be applied to the heat exchange coefficient and to the critical heat flux as calculated by the current version of the CATHARE2 code. Best-estimate calculations taking into account the point kinetics model of the code and the estimated corrective factors were performed. These calculations showed that the tests with high reactivity insertion are better predicted. Nevertheless there are still some code-to-data discrepancies for the transients with low reactivity insertions.

It appears that the moderator feedback effects reported in the benchmark specifications (measured in specific steady state conditions at homogenous temperatures in the core and in the pool) were not appropriate for the transient tests. Additional neutronics calculations should be performed to estimate the moderator feedback effect in the core during the transient tests. These coefficients will be introduced in the kinetics model of the CATHARE2 code and the tests will be recalculated.

Ghana

The Ghanaian team analysed two reactors: SPERT-III and MNSR-Y. Two codes were utilized in for the two reactors and these are the MCNP5 for neutronics and PARET for Steady State and Transients Analyses. For the MNSR-Y, the data provided by the Syrian counterpart was used to prepare the input file for MCNP5 computations. Parameters calculated include multiplication factor, delayed neutron fraction and reactivity feedback coefficients. The results from the PARET computations do not agree very well with results of the other counterparts in the group. There is the need to re-check our analysis. For SPERT-III, the parameters were provided by the USA. The consolidator of the group is yet to compare our results with the two other counterparts. Hence, further work will depend on recommendations and comments put forward by the consolidator. Both MCNP5 and PARET have been used extensively for the Reactor Core Conversion Analyses for Reduced Enrichment Program and they have proved to be reliable with our experimental data.

Argentina

Commitments

Three different facilities were modeled and analyzed for several benchmarks using RELAP5 mod 3.2, they are:

✓ **ETRR-2 (Egypt)**

- Several steady state operations
- Loss of Flow transient
- Negative reactivity insertion

✓ **RSG-GAS (Indonesia)**

- Loss of Flow transient

✓ **IEA-R1 (Brazil)**

- Core configuration #243: Several steady states and Loss of Flow transient

- Core configuration #246: Steady states with and without a box around the core and Loss of Flow transient with and without box around the core

Individual and general reports have been sent to D. Ridikas.

Comments on results

In general, steady-state results were properly reproduced except for the IEA-R1 facility where the inlet and outlet core coolant temperatures are in good agreement, but the cladding temperatures present a difference of several degrees above the measured values while coolant temperatures in the instrumented fuel assembly are underestimated.

In every Loss of Flow transient simulation, the coast-down flow shows a very good agreement with measurements.

ETRR-2: The agreement between measured and calculated data is acceptable for the first 100. seconds of the transient. When 3D effects inside the chimney become relevant the discrepancy of results becomes larger.

It means that RELAP (when used as a 1D model) is not the appropriate tool to reproduce this 3D phenomenon at least with the available information.

RSG-GAS: Results are in good agreement with measurements. RELAP can predict properly the trend of coolant and cladding temperatures when recorded in coolant channels (1D geometry). The absolute values are higher than the recorded ones, resulting in a conservative approach.

IEA-R1: Calculated cladding temperatures present a conservative difference of several degrees when compared with measurements. On the contrary, coolant temperatures in the coolant channels are underestimated. These differences could be due to the contact of thermocouples, whether it is good or not, or that measurements are performed on a single fuel plate while calculations correspond to a lumped control volume of six fuel plates, according to data supplied by the owner of the facility.

Syria

Syria participated in the benchmark analysis of five facilities. The activities cover the benchmarking of both neutronic and thermalhydraulic tools. Codes used were: MCNP, CITATION, MERSAT, PARET and RELAP (see table below). Besides, the Syrian team was responsible for establishing the data base for the MNSR reactor. During the last year of the CRP the Syrian CSI was responsible as consolidator for benchmark analysis of two facilities: IEA-R1 and MNSR.

Activities and Achievements of Syrian team during the 3red year of CRP

Reactor	Task	Code	Remarks
1. ETRR2	TH: SS, LOFA, and RIA	MERSAT, RELAP	Benchmarking and code-to-code comparison
2. RSG-GAS	TH: SS, LOFA	MERSAT, RELAP	Benchmarking and code-to-code comparison

3. IEA-R1	TH: SS, LOFA	MERSAT, RELAP	Benchmarking and code-to-code comparison
4. SPERT-IV	Neutronic: $k_{eff,}$ criticality	MCNP	Benchmarking and code-to-code comparison
	TH: RIA	MERSAT, RELAP PARET	Benchmarking and code-to-code comparison
5. MNSR	Neutronic TH	MCNP, CITATION, WIMS MERSAT, RELAP	Benchmarking and code-to-code comparison

General Recommendations to the Project Outcomes

- The project aims at enhancing national capabilities for RR safety analysis applicable to reactor design and operation. The successful project implementation shows that these objectives have been fulfilled in view of the number of actively participating teams, facilities and codes being applied in the benchmark analysis.

- The benchmark results show that the employed neutronic and thermalhydraulic tools are very useful for performing deterministic safety analysis of RR. However, further code improvements and validations is still required. Especially to capture 3-D Thermal-Hydraulic effects.

- IFE is a useful integrated effect test for code validation. However, high quality of experimental data is requested to base upon for code validation.

- The use of the RELAP code by different analysts for the same benchmark calculation and the observed discrepancy in the achieved results identifies the importance of “user effect.” This effect can be minimized by enhancing the communication between users. A web-based link between users could assist in establishing general user guides.

- SPERT-IV experiments prove to be very useful for code validation due to their variety and importance for RR safety. We plan to complete working on this set and exchange experience between interested teams beyond the CRP.

South Africa

OPAL summary:

For the OPAL benchmark problem South Africa submitted 3 sets of calculated results, using three different methods; nodal diffusion (OSCAR-4); Monte Carlo (MCNP) and a hybrid between Monte Carlo and deterministic methods (CUCGP).

For the compact core geometry of the OPAL reactor, the OSCAR-4 system had difficulties calculating some of the required results to the desired accuracy. In particular new developments had to be made to be able to place two control rods within a single node in the model.

The CUCGP results showed deviation from the experimental results. This methodology is still in the development stage and the benchmark shows there is room for improvement.

The MCNP results showed good agreement with the measured data.

MNR summary:

Calculations for the control rod calibrations, radial flux wire mapping and axial flux wire measurements have been completed and submitted. Some final requests from the data provider have been received to compile all the results, including what was decided during the course of the meeting, into a final coherent set for review. Further investigations have been identified, but will be carried out outside of the scope of the CRP.

Some issues with the provided data have been identified, which led to small modifications to the points on which the results will be compared. In particular, for the control rod calibration experiments, measured reactivity worth was not provided. Also, no specific procedure for determining the total worth of each rod was specified which lead to differences in the experimental results from each of the participants.

Romania

The status of the:
- SPERT3 benchmark is 95% completed. Per the request of the group consolidator formulated during the meeting, peak cladding temperature and compensated reactivity for the analyzed series of tests will be submitted during January 2013;
- MINERVE benchmark is 95% completed. France (CEA), as a group consolidator, suggested reviewing the possible sources of discrepancy in modeling the Valmont pin. If motivating issues will be discovered, probably a few more MCNP runs are still needed to improve the accuracy of the results. The same for axial power distribution in what concerns the number of histories in the MCNP calculation.

There is another benchmark for which we performed calculations without an official commitment: the IEA-R1 loss of flow test. The completion is uncertain, as well as the quality of experimental data provided, as resulted from the open discussion on Monday. If continuation of work by Romania is really needed, specific requirements should be formulated. At the same time, the next months of work are dedicated to finalizing the above two benchmarks and, should the need for IEA-R1 be clearly expressed it will require a more extended deadline. Even if not really needed, I suggest a thorough scrutiny of the thermocouple data, with formulation of resulting conclusions inside the CRP document.

Korea, Republic of

In the 3rd CRP meeting, KAERI made a commitment to accomplish thermal hydraulic analyses of the IEA-R1 experiments using RELAP5/MOD3 and neutronic analyses for the OPAL experiments using McCARD.

KAERI completed the commitment and submitted the calculation reports prior to this meeting. The chief scientific investigators presented and discussed the calculation results.

The RELAP5/MOD3.3 simulation of the IEA-R1 experiments reached the following conclusions:

(1) The coolant and fuel cladding temperatures calculated by the present RELAP5/MOD3.3 modelling follow well the overall trend of the measured temperatures.
(2) However, the calculated coolant and fuel cladding temperatures increase and decrease much faster than the measured temperatures during the loss of flow test.
(3) The calculated peak temperatures of the coolant and fuel cladding are generally higher than the measured peak temperatures during the loss of flow test.
(4) These discrepancies are expected to be caused from the one-dimensional modelling of the coupling valve where the complicated three-dimensional flow is expected.

From the comparison of the RELAP5/MOD3.3 calculations and the IEA-R1 experiments, the following recommendations are made:
(1) More accurate thermal power of the IFA and flow rate through the IFA during steady state operation should be considered in simulation.
(2) The detailed dimensions of the reactor structures should be provided to calculate the coolant volumes and pressure loss coefficients accurately.
(3) The measured coastdown flow of the primary cooling system should be clarified and understood by the analysts.
(4) Uncertainties on the measurement of the coolant and cladding temperatures should be analyzed. The effects of thermocouple disks with a size of 10 mm in diameter and thermocouple lead wires on the thermal hydraulics in the narrow channel and local heat flux distortion should be taken into consideration in the uncertainty analysis.

The McCARD simulation of the OPAL experiments is summarized as follows:
(1) Steady state flux distribution
The calculated thermal fluxes are normalized at a core power of 42 kW. The thermal neutron flux distribution by McCARD calculation shows good agreement with the experimental results within 6% error except several points in all cases. The McCARD predicts well the measured thermal fluxes in the core region of the OPAL reactor.

(2) Control rod worth
The McCARD code predicts well the criticalities within 170 pcm error. The maximum difference between measured and calculated integral control rod worth was about 6 %. The differential control rod worth is not compared due to its small value of which prediction requires very long computation time in Monte Carlo calculation. McCARD predicts well the criticality for critical core configuration and the control rod worth.

(3) Kinetic parameters
In the McCARD calculation, kinetics parameters such as β, ρ and Λ are provided in the output file. Using these kinetics parameters, the value of α can be easily obtained. The calculated α is 38.96 s^{-1}, and agreed well with the experimental value within 2% error.

(4) Fuel burnup
We did not complete the analysis yet.

Canada

Canada's (CAN) contribution to the IAEA CRP 1496 involves both data consolidation and analysis results submission.

Results for selected sections of the MNR Benchmark Problem were provided for comparison to experimental measurement and to other group submissions.

Data consolidation was performed for both the MNR Benchmark Problem (in collaboration with the SAF NECSA group), and the Static section of the SPERT-IV D-Core Benchmark Problem. For both problems, the facilities and benchmarks were reviewed, the consolidated results were presented and preliminary conclusions were summarized. For the MNR Benchmark problem there were sections of the problem which were only completed by a single participant. In these cases the sections were not included in the data consolidation. Preliminary conclusions and recommendations were compiled. Recommendations for improvements in the experimental aspect of this problem were compiled and presented. Discussions with the participating groups were continued in light of the preliminary data consolidation, final requests were made, and comments/findings will be included in the final individual reports and the data consolidation report.

With regards to the SPERT-IV benchmark problem; data consolidation results, conclusions, and recommendations were presented and discussed for the Static sections of the problem. Final requests for information were made to all participants and comments will be incorporated into the final consolidation report. Data consolidation was not completed for the flux wire distribution section of this problem due to inconsistent approaches used by the different groups on normalization and averaging. This is to be standardized and results to be resubmitted. Canada also participated in the concluding discussion for the Transient section of the SPERT-IV problem providing various suggestions and requests of the participants.

Egypt

A summary of current and pending status
Participant : Ibrahim D. Abdelrazek, Egypt

Provision of Data

Since the start of the CRP activities, the following documents were provided:

1- ETRR-2 reactor facility specifications
2- ETRR-2 physics data
3- ETRR-2 Experimental results (steady state, LOFA and NRI)

Contractual commitment

- Thermal hydraulic calculations for ETRR-2 reactor (steady-state temperature, transient analysis : LOFA and reactivity insertion (RI));
- Steady-state thermal hydraulic calculations for RSG-GAS reactor;
- Analysis of WWR-SM reactor: Steady-state and reactivity insertion analysis;
- SPERT-IV: Steady-state and reactivity insertion analysis.

Status

All the workload, with the exception of SPERT-IV thermal-hydraulics, has been done. The final report which was submitted, along with the excel sheets, include the description of the facility, experiments, results as well as the results and conclusions for each facility. For the case of SPERT-IV, however, as RELAP5 is being used to simulate the facility, further work has to be done to finalize the calculations. To complete that task 6 more weeks are needed.

USA

Remarks concerning the Benchmark Specification

Detailed Specifications created by USA are given in the companion document: IAEA CRP: Innovative Methods for Research Reactors, SPERT-III E-CORE Reactor Specification, IAEA, 2012.
Early on there we a few requests for clarification of items in the benchmark specification, but none in the past year. Romania was able to proceed with their analysis without any need for more information. Their success in producing good agreement with measurement indicates that the specification was complete.

Remarks Concerning Analysis of the Experimental Program

The E-core experimental program was divided into low-initial-power and high-initial-power test phases. Low-initial-power (≈50 W) excursions were performed for cold- and hot-startup conditions. High-initial-power excursions were performed for hot-standby and operating-power conditions. Reactor physics/thermal hydraulic analysis was performed for three different reactor conditions of temperature, pressure, coolant flow rate, and initial power.

Comments on estimated standard deviation for measured parameters

- Reactor period 2 %
- Reduced prompt neutron generation time 2.5 %
- Delayed neutron parameters 7-15 %
- Derived reactivity insertion 4 %
- Reactivity compensation at peak power 11 %

We suspect that the reactivity insertion was actually known more accurately that the quoted standard deviation of 4% because the effect of this apparently small range is so huge. As a result, Romania will perform a sensitivity study at ± (1/2) 4% for case T-86 and will include that study in their final report.

Comments on the MCNP model for neutronics

The MCNP5 code used for the analysis by USA was version 1.60, with standard libraries (ENDF-B/VII). It is documented in LA-UR003-1987, MCNP— A General Monte Carlo N-Particle Transport Code, Version 5, Volume I: Overview and Theory, X-5 Monte Carlo Team, April 24, 2003 (Revised 2/1/2008). This version has the ability to calculate point kinetics parameters directly. Comparisons have been made recently at ANL for another reactor, between the new option, and traditional methods of obtaining beff etc. This camparison validated the new option. We believe that our results are correct.

Direct heating to the moderator was not calculated. Instead, a value of 2.6% was assumed as typical of a PWR UO2 fuel rod [Shigeaki Aoki, Takayuki Suemura, Junto Ogawa and Toshikazu Takeda, Analysis of the SPERT-III E-Core Using ANCK Code with the Chord Weighting Method, Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 46, No. 3, p. 239–251 (2009)]. Subsequent PARET calculations for experiment T-86 confirmed that direct heating was significant. As a result, experiments T-79 through T-86 were recomputed using direct heating. The other cases assumed no direct heating, because the coolant temperature rise in those tests was so small as to eliminate the effect of direct heating.

Comments on The PARET model

Analysts at ANL are divided as to what is the best procedure to follow when creating a PARET model. Some believe that a two-channel model is best when one only has reactor-averaged feedback coefficients. In that case, one would use one channel to represent the hottest fuel rod or plate, and the

2nd channel to represent the remainder of the core. USA created a 5-channel core representation in order to attempt to follow the consequences of heatup of smaller groups of channels, rather than one representing the core average. The 5-channel model would be even better if channel-dependant feedback coefficients were available (this requires much more analysis).

Status of Reports

The existing report by Romania is complete regarding graphics for a large number of tests. It is incomplete regarding their methodology. The have committed to providing results for peak clad surface temperature (the peak temperature measured at any time during the event). Also, the amount of reactivity feedback at the time of peak power was part of the output but not yet inserted into the report. The existing report by USA has covered all the planned tests and more. Graphics comparing calculation with experiment are provided. Peak clad surface temperature rise values are provided. It is concluded that reactivity compensation at peak power is good. USA also will review their work during January to check Doppler feedback calculations and control rod worths. USA also will request ANL approval for release of their report to the IAEA.

Discussion of Results Concerning Reactivity Feedback, and Clad Surface Temperature Rise

There is some uncertainty as to the initial power of each test. One can see that typical calculations of power show about the same slope, indicating that the period is correct, but that there may be a time offset caused by this uncertainty in initial power.

It is noted that the PARET results are always conservative: they predict too high a peak power, and too high an energy release. This leads to predicting too high a temperature rise in the clad. This comparison is somewhat imprecise in that the axial locations of the measurement may not be quite the same as computed, and because the axial power shape in the calculations is sensitive to the position of the control rods. It is concluded that the Cold-Startup tests, which had no flow at the start of each transient, is seriously over-predicted. The other test conditions with flow also over-predict, but appear to be quite reasonable.
One can also observe that the trends in power vs. reactivity are excellent. This is fine for reactor safety and licensing because then the reactor performance can conservatively be predicted for similar designs, for similar conditions covered by the test envelope.

Clearly, the Doppler effect from heat-up of the UO_2 dominates the shape of each test power vs. time curve. There is so little temperature rise in the low-power tests that there is no void production, and the temperature coefficient for the water is quite small.

Pakistan

SPERT-IV was selected for benchmark calculations. Neutronic analysis was performed for the D-12/25 core. The D-12/25 core was the first core installed in SPERT-IV. The SPERT-IV facility was a pool type system with provisions both for forced coolant flow and 1 MW-capacity heat removal. This core was composed of 25 fuel assemblies in a square five by five section of the nine by nine supporting grid. Design data and fuel elements specifications of the SPERT-IV reactor were taken from the supporting document “SPERT-IV D-12/25 Reactor Specification” provided by IAEA.

The core parameters calculations involved two steps; i.e. cross-sections generation and core integral parameters calculations. Computer codes WIMSD and CITATION were employed.

Flux profiles were calculated at various locations. There is good agreement between the calculated and the experimental data in the lower regions of the core i.e. in the regions where control rods have been withdrawn. However there is a maximum of about 10% deviation between experimental and calculated results. Various core characteristics, namely criticality position, excess reactivity, shutdown margin, control rod worth, reactivity coefficients, and reduced prompt neutron life time were also computed and compared. The pending results of the analysis shall be communicated to the consolidator as per deadline made in the meeting.

MNSR-Y was selected for performing neutronic analysis. It is the Syrian MNSR (Miniature Neutron Source Reactor). Standard nuclear reactor codes WIMSD and CITATION were employed. WIMSD was employed for macroscopic cross-section generation while core modeling has been performed in CITATION. WIMSD uses 69 groups, multi-region integral transport theory to solve the neutron transport equation for the lattice cells. CITATION uses a finite difference scheme to solve the neutron diffusion equation in one, two and three dimensions.

Parameters checked are *core effective multiplication factor, criticality position of the fresh reactor, control rod worth, thermal flux values (both at outer and inner sites), power densities, moderator temperature and moderator density coefficients of reactivity, radial and axial power distribution (at 30 kW)*. Comparison of different calculated and quoted values of control rod worth has been made. The pending results of the analysis shall be communicated to the consolidator as per deadline made in the meeting.

Indonesia

Following the presentations by the consolidator for the calculationsal results by participants, it became apparent that there are some important data which were not provided from the experimental results/reference. Examples:

1. specific heat
2. thermal conductivity
3. coolant velocity in the core

Thermocouple position data provided are inconsistent (from reference). Then, the axial power peaking factor was averaged for the two positions provided of instrumented fuel. This can lead to some discrepancies to the analysis results.

Therefore, for the thermalhydraulics benchmarks, the above parameters should be provided more accurately. If adequate data is provided, the user effect can be minimized.