Current Development and Trends In Thermal-Hydraulics

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Abstract. A review of CSNI activities during the last two decades in the field of thermal-hydraulics and related topics has been extensively presented in sessions II to IX. New activities are in progress or planned partly based on recommendations of the CSNI Operating Plan and the CSNI SESAR SFEAR report, but also on requests coming from the member states. These activities are performed in the frame of the CSNI Working Group on the Analysis and Management of Accidents (GAMA) or in the frame of CSNI Projects. These actions are summarized in this paper.

1. INTRODUCTION

In Sessions II to IX the most significant achievements of CSNI activities in the field of thermal-hydraulics have been presented. On-going and planned actions are closely related to the CSNI Safety Issues and Topics (SIT), which are revised from time to time. Based on a recent review of the SITs, the ones having a close relation to thermal-hydraulics, along with the main CSNI/CNRA challenges, are listed below:

1. **Shrinking nuclear infrastructure**
   a) Knowledge Management (… The explicit and implicit knowledge residing within this workforce should be captured and made available to the future workforce, as done for instance in THICKET. Maintaining data is part of knowledge management…)
   b) Experimental Facility Loss (… An adequate and cost-effective set of facilities should be maintained to address emerging safety issues…)

2. **Increased public expectation on safety in use of nuclear energy**
   b) Transparent technical basis for safety assessment (Building an international perspective and consensus in addressing nuclear safety issues, through for instance state of the art reports or workshops…)

3. **Industry initiatives to improve economics and safety performance**
   b) Maintaining Safety Margins (There are considerable economic incentives to introduce new operational strategies and advanced, best-estimate tools for safety assessments of nuclear facilities. Adequate safety margins need to be maintained as these changes are made…)

4. **Necessity to ensure safety over plant lifecycle**
   b) New Risk Perspective and Safety Requirements (…Each Working Group is expected to assure that new information, experimental data, improved evaluation tools are properly taken into account.)

5. **New reactors and new technology**
   c) New concepts of operation (Modular reactors, passive safety features, advanced control rooms and different workforces are all elements of new design that challenge traditional assessment methods. New methods may need to be developed to address the safety implications of these advancements…)
   d) New methods and tools (New analytical techniques such as computational fluid dynamics and virtual reality simulators are elements of new designs that challenge traditional assessment methods. New methods should be developed to address the safety implications of these advancements…)

Taking into account the listed SITs the CSNI and GAMA present and planned activities in the thermal-hydraulic field focus on safety analysis methods in order to check or improve:

- The prediction of accidental plant behaviour by means of codes (postulated accident or accident management actions)
- The evaluation of the codes (physical models, code assessment, experimental programs)
- The evaluation of the methods used in the code application (conservative, best estimate, uncertainty evaluation,…)
- The evaluation of the safety margins of accident analysis

The evaluation of the acceptability of the results and eventually of the accident management actions in relation with safety criteria. These activities are reviewed in the present paper.

2. COMPUTER CODES

As far as evaluation and assessment of thermal-hydraulic systems codes is concerned most of the present activities is carried out in conjunction with the experiments performed within the OECD projects: this will be discussed in Section 3. Major effort in the GAMA group with respect to codes is related to activities, which will be continued in the coming years: assessment of CFD codes for nuclear safety applications, best estimate safety analysis methods including uncertainty evaluation and organisation of International Standard Problems.

2.1. Assessment of CFD Codes

As it was presented in Session IX, important activity is going on within GAMA to assess the role and the maturity of CFD codes in nuclear safety. As regards future activities in the field at the last GAMA meeting in September 2007 it was suggested to form a group with the mandate to identify and prioritise the needs for future CFD work and propose a work scope for GAMA in the CFD area. The group drew up a list of country-specific safety items for which CFD was considered to bring real benefits. A synthesis of all information collected enabled an overall priority ranking of the safety items. Due to the fact that the level of maturity of CFD codes for single-phase and multi-phase problems shows important differences, the ranking was performed separately, as presented in Tables 1 and 2.

One of the future activities that can be proposed is the organisation of benchmark exercises for the high priority issues. This obviously needs experimental data to be compared with calculation results: for the single phase problems these are more or less available (with the exception of aerosol deposition in containments, since comprehensive, local aerosol deposition data appear only to be available for pipes). Though CFD modelling is much

Table 1: Ranking of Single-Phase CFD Issues.

Topic (single-phase)	Score/36	Generic Interest
PTS	31	HIGH
Hydrogen mixing and combustion in containments	30	
Flows in complex geometries*	29	
Boron dilution	27	
Sump strainer clogging	26	
Aerosol deposition in containments	25	
.	.	.
Thermal fatigue	23	MEDIUM
Hot-leg heterogeneities	21	
MSLB (leading to asymmetric flow)	20	
HTGR lower plenum mixing	16	
HTGR core heat transfer	15	
.	.	.
HTGR reactor cavity cooling heat transfer	13	LOW
GCR/VHTR heat transfer issues	13	
Flow behind blockages in LMFRs	9	
Flow-induced vibrations in LMFRs	8	
Core barrel vibration in APWR	6	
Special issues for CANDU reactors	3	

Table 2: Ranking of Multi-Phase CFD Issues.

Topic (multi-phase)	Score/36	Generic Interest
PTS	27	HIGH
Reflooding/following LB-LOCA, including UPI and EPR	27	
CHF	26	
Condensation-induced water hammer	26	
.	.	.
Sub-cooled boiling in PWRs	23	MEDIUM
Steam condensation in pools	19	
Induced break	15	
.	.	.
Gas entrainment in LMFRs	9	LOW
Special issues for CANDU reactors	3	

less mature for two-phase flows, two high-priority safety items were identified for which it was considered the effort of benchmarking was worthwhile: PTS (above-surface injection in a partially-filled pipe) and reflooding following a LB-LOCA.

Following the CFD4NRS workshop held in Garching in 2006, a new one, XCFD4NRS is being organized, taking place in Grenoble, this autumn. This Workshop is intended to extend the forum created for numerical analysts and experimentalists to exchange information in the field of nuclear reactor safety (NRS) relevant to CFD code validation, but this time with more emphasis placed on new experimental techniques and two-phase CFD applications.
An additional activity that has already started within GAMA consists in designing and keeping up-to-date a web-based portal, disseminating information on the usage of CFD codes for NRS applications, based on the results obtained in previous CFD-related activities. The information will include the following:

- Best Practice Guidelines for the use of CFD in NRS applications
- Assessment of CFD codes for single-phase NRS applications
- Assessment of CFD codes for multi-phase NRS applications.

2.2. Best Estimate Codes Including Uncertainty Analysis

As discussed in Session VI, the practicability, quality and reliability of BE methods, including uncertainty evaluation in applications relevant to nuclear reactor safety has been assessed in the BEMUSE project, with the aim of making recommendations regarding BE methods and tools, and their application in the licensing process. Since the last two phases of the BEMUSE project are still running, final recommendations can be expected only by the end of 2008. In the mean time the GAMA took the initiative to enquire about the member countries' needs in the field. In a first step a questionnaire was distributed to the regulatory bodies of the member states with the following questions:

1. Please provide a summary of applications to the Regulatory Body in your country making use of a best-estimate methodology. Please, include the safety case, the name of the computer code and of the uncertainty method applied (possibly with references).

2. What are the potential problem areas in your country regarding the use of best-estimate methodology in safety analysis, such as:
   - Selection of an appropriate uncertainty methodology,
   - Definition of the uncertain parameters,
   - Definition of uncertainty distributions for the parameters,
   - Other.

3. Which areas should GAMA cover in the future in order to complete or extend the current BEMUSE work, such as:
   - Generic methodology assessment,
   - Application to specific reactor safety issues,

- Application to licensing, e.g. for power up-rates,
- Other.

4. Which type of activity should GAMA cover in the future in order to complete or extend the current BEMUSE work, such as:
   - International Standard Problems
   - Comparative short-term analyses of specific events/transients
   - State of the art reports
   - Other

Although as of today only a limited number of responses have been received, the following activities seem to be endorsed in the future:

- Identification of the uncertain parameters and related distributions
- Application of uncertainty methods to CFD or 3D codes
- International Standard Problems (focus on separate effects test rather than integral one, as in BEMUSE: CEA proposes a PERICLES reflood test)
- Benchmark for the determination of the uncertainties of the input parameters
- Comparative short-term analyses of specific events/transients

Also the OECD/NEA Nuclear Science Committee (NSC) created an Expert Group on Uncertainty Analysis methods in Modelling, which addresses multi-scale / multi-physics aspects of uncertainty analysis and will work in close co-ordination with the benchmark groups on coupled neutronics-thermal-hydraulics simulations and on coupled core-plant problems. The Expert Group will also coordinate its activities with CSNI GAMA. The Expert Group has the following mandate:

1. To elaborate a state-of-the-art report on current status and needs of sensitivity and uncertainty analysis in modelling, with emphasis on multi-physics (coupled) and multi-scale simulations.
2. To identify the possibilities of international co-operation in the uncertainty analysis area that would benefit from coordination of the NEA/NSC.
3. To create a roadmap for the development and validation of methods and codes required for uncertainty analysis including the definition of benchmarks.

2.3. International Standard Problems

A long series of ISPs were organised in the past under the umbrella of OECD CSNI. Most of these were related to system thermal-hydraulics, as it was discussed in detail in Session V, with the aim to increase the confidence in thermal-hydraulic systems codes. As these codes reached a fairly high level of maturity by the end of the last century, ISPs in system thermal-hydraulics haven't been proposed in GAMA, and interest moved towards severe accident and containment analysis. To some extent, the first part of the BEMUSE activity, the reanalysis of the LOFT L2-5 LBLOCA test can be considered as an ISP: in fact, one of the aims of this activity was to demonstrate the improved capabilities of systems codes as compared to ISP 13, run more than 20 years ago.

A number of benchmark activities launched by the NEA Nuclear Science Committee (NSC) should also be mentioned in this context. Traditionally, NSC's main focus is on neutronics and – since coupling between 3D neutronic and systems thermal-hydraulic codes became state-of-the art in the past decade – NSC was proposing several benchmarks for coupled calculations.

The first one was the PWR Main Steam Line Break (MSLB) benchmark, completed in 2002. Based on real plant design and operational data for Three Mile Island Unit 1 Power plant, it was divided into three phases:

- Phase 1: Thermal-hydraulic system modelling of the MSLB transient with specified point kinetics parameters;
- Phase 2: Coupled 3-D neutronics/core thermal hydraulics evaluation with imposed thermal hydraulics boundary conditions;
- Phase 3: Best Estimate coupled core-plant transient analysis.

Following the same three phases approach, two others benchmarks have been initiated:

- The BWR Turbine Trip Transient benchmark with the objective to analyse complex transients with coupled core/plant interaction and validate Best Estimate code predictions with experimental data from Peach Bottom Unit 2; this benchmark was completed in 2005.

- The VVER-1000 Coolant Transient benchmark with the objective to validate coupled 3-D neutron kinetics/thermal-hydraulics system codes for application to VVER-1000 reactors based on actual plant data. Two types of transients have been proposed:
- V1000CT-1: switching on one main coolant pump with the other three in operation (completed last year)
- V1000CT-2: coolant mixing experiment and main steam line break scenario (started in 2006)

The above described NSC initiated activities heavily involve experts in thermal-hydraulics as well: this was another reason, why thermal-hydraulic ISPs were not launched within GAMA in this decade.

There is a new challenge for thermal-hydraulic systems codes: it should be assured that aspects of new power plant designs are properly handled by them. In view of this challenge it was decided in the last GAMA meeting to launch an ISP based on a brand new, large scale test facility in Korea.

Korea Atomic Energy Research Institute (KAERI) has recently started the operation of a new thermal-hydraulic integral effect test (IET) facility for pressurized water reactors - ATLAS (Advanced Thermal Hydraulic Test Loop for Accident Simulation). ATLAS is designed to simulate transient and accident conditions of the major reactor types of Korea, APR1400 and OPR1000(KSNP). In particular, ATLAS will provide unique test data for the 2(hot legs) x 4(cold legs) reactor coolant system with direct vessel injection (DVI) of emergency coolant; this will expand significantly the currently available data bases for code validation.

The ATLAS facility is designed as a half-height facility by applying Ishii's three-level scaling methodology, which consists of integral scaling, boundary flow scaling and local phenomena scaling.

Major design characteristics are based on the APR1400 reference reactor:

- 1/2 height and length, 1/144 cross-sectional area (1/12 diameter), 1/288 volume
- 2(hot legs) x 4(cold legs) reactor coolant loops and integrated annular downcomer
- Prototypic pressure and temperature conditions
- Direct vessel injection (DVI) of emergency core cooling (ECC) water
- Max. 10% of scaled power (Core power: 2 MW)

with incorporation of OPR1000 design features, e.g. cold leg injection capability and low pressure safety injection system.

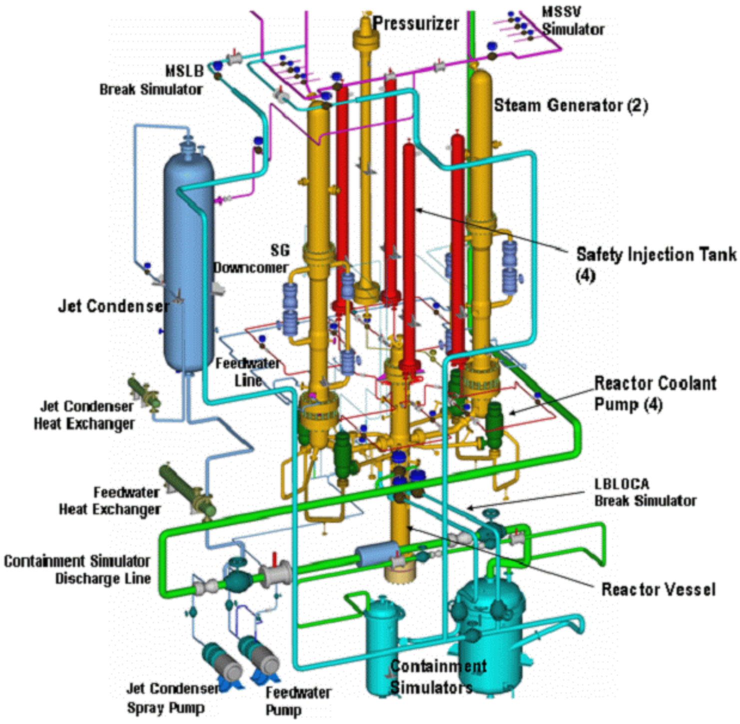

Figure 1: Major Components of the ATLAS Facility.

The purpose of the proposed ISP exercise is to perform blind and/or open calculations by participating organizations for a unique thermal-hydraulic integral effect test in ATLAS. The test scenario will be specified based on agreement among the participating organizations. One of the following tests is proposed by KAERI:

- a double-ended guillotine break in one (1) of the four (4) direct vessel injection (DVI) lines
- a small- or intermediate-size cold leg break with DVI of ECC water.

Final decision on the selected scenario will be taken at the next GAMA meeting in September 2008 and the ISP is planned to start in 2009.

3. OECD PROJECTS

As a result of the shrinking nuclear infrastructure in the past decade an important number of thermal-hydraulic facilities had to be closed down world-wide. Recognizing the danger associated with this development one of the CSNI safety issues states: “An adequate and cost-effective set of facilities should be maintained to address emerging safety issues”. This action requires definition of meaningful experimental programmes for the supported facilities and their fulfilment in the form of OECD projects. Although the primary aim of these projects is to produce test results, in almost all the cases this is coupled with an extensive code validation work within the project.

3.1. The ROSA Project

The project was started in 2005 with the aim to resolve PWR thermal-hydraulic safety issues through LSTF experiments. ROSA/LSTF of Japan Atomic Energy Research Institute (JAERI) is an integral test facility designed to investigate thermal-hydraulic response during PWR transients and accidents. The facility simulates a 1000 MWe four-loop Westinghouse-type PWR with full-height and two equal loops at 1/48 volumetric scaling. Experiments can be performed in a wide range of system pressure, i.e. from full pressure to atmospheric pressure. The experimental programme focuses on the verification of models and simulation methods for complex phenomena that can occur during reactor transients and accidents and is divided in the following test groups:

1. Temperature stratification and coolant mixing in horizontal leg & annular downcomer during ECCS coolant injection
2. Unstable and destructive phenomena: water hammer in horizontal legs under wide range of system pressures
3. Natural circulation under high core power conditions, possibly with CCFL at SG U-tube inlet, fast boil-off and temperature excursion in the core
4. Natural circulation with super-heated steam in BDBA conditions
5. Primary cooling through SG secondary depressurization with non-uniform parallel channel flows among SG U-tubes, with and without non-condensable gas
6. Open subjects defined by the participants:
   1 & 2: pressure vessel upper-head break and bottom break LOCAs, coupled with symptom-oriented operator actions
   3: steam condensation on ECCS coolant in cold leg during LBLOCA

All experiments with the exception of Nos. 4 and 6.3 have already been carried out, with the project activities to be closed in 2009. The project partners have performed a large number of computer code analyses, mostly with systems codes but in the case of Test 1 CFD codes have also been applied (and this is to be expected for Test 4 as well). A follow-up ROSA project is under discussion – for the time being among the present project partners.

3.2. The PKL Project

PKL is a large-scale integral test facility with 4 primary simulating a typical western-type 1300 MW PWR. Modelling of a 3-loop plant is possible by simply isolating one loop. This allows a realistic simulation of accident transients under symmetric and non-symmetric conditions. Volume–power scale of PKL facility is 1:145, while elevation scale is 1:1.

The final report of the first PKL project was released recently and – due to the general interest of the subjects treated – it is proposed to publish it as a CSNI report. The project was focusing on two items: boron dilution and specific aspects of residual heat removal (RHR).

The tests run to investigate the inherent boron dilution process during a SBLOCA led to the following conclusions:

- Formation and accumulation of low borated water was experimentally confirmed, but accumulation of low borated water occurs only at significantly reduced primary water inventory.
- The maximum size of low borated water slugs was smaller than expected and effective mixing takes place in the loops and in the steam generators.
- Displacement of low borated water slugs into the reactor pressure vessel does not occur simultaneously in the individual loops.

The tests focusing on loss of residual heat removal identified new phenomena, such as formation of active and passive heat transfer zones in the SG U-tubes, subcooled water columns and nitrogen blocking the circulation in some of the U-tubes and spillover of water from the SG inlet to the outlet side in others.

The PKL test results were extensively used for code validation within the project: three analytical workshops dedicated to PKL experimental results were organized. Pre- and post-test calculations, sensitivity studies, plant analyses were performed by 15 organizations using the following codes: RELAP 5, CATHARE, ATHLET, MARS, TRACE, APROS.

The PKL team proposed to investigate the still open question with respect to loss of RHR in a new project. The PKL-2 project will have its kick-off meeting in June 2008 and the 3.5 year project covers eight integral experiments in PKL with the following topics:

- Heat transfer mechanisms in the SGs in presence of nitrogen (complemented by tests in the PMK test facility for horizontal SG)
- Accident situations under reflux condenser conditions for new PWR design concepts
- Fast cooldown transients such as main steam line break (with complementary tests in the ROCOM test facility on mixing in the RPV downcomer and the lower plenum)
- Boron precipitation processes after LB-LOCA

The subjects of 2 tests are still open and will be defined during the programme period in agreement with the project partners.

3.3. The PSB-VVER Project

The project is described in detail in Paper 5 of the meeting. It should, however, be mentioned here that the test results of the project have been used by the partners to assess the capabilities of both Western and Russian codes to describe VVER-specific phenomena.

4. REGULATORY ISSUES

It is in the mandate of GAMA to assess and strengthen the technical basis needed for the prevention, mitigation and management of potential accidents in nuclear power plants, and to facilitate international convergence on safety issues. In that context it endeavours to provide answers as requested by CNRA or regulatory bodies represented in GAMA. Two activities may be mentioned here.

4.1. Sump Strainer Clogging

The issue is related to the performance of the ECCS in the recirculation mode: it is dependent upon the ability of the containment sump strainers to remove debris without plugging to a point, which would reduce recirculation flow to a dangerous level. This includes consideration of solid debris as well as chemical effects, which can cause gelatinous material. The problem was raised in 1992 by an event at Barsebäck Unit 2, a Swedish BWR, where the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray system suction strainers with mineral wool that required operators to shut down the spray pumps and backflush the strainers.

A series of NEA activities followed, resulting in reports and workshops:

- Report: Knowledge Base for ECCS Recirculation Reliability, NEA/CSNI/R (95)11
- Workshop on Update of the Knowledge Base for Sump Screen Clogging, Proceedings, May 1999, Stockholm, Sweden, NEA/CSNI/R(2000)16

- Report: Knowledge Base for Strainer Clogging - Modifications performed in different countries since 1992, NEA/CSNI/R(2002)6
- Workshop on Debris Impact on Emergency Coolant Recirculation; Albuquerque, NM, USA, 26-27 February 2004, NEA/CSNI/R(2004)2

In the last CSNI meeting a proposal was put forward to set up a permanent website on the sump clogging issue containing event and phenomena descriptions, experiments, regulatory approaches and requirements. Considering the diverse conditions and configurations of different power plants, the CSNI recommended presenting and discussing the proposed activity to GAMA at the September 2008 meeting. A CNRA workshop is planned in the subject for December 2008.

4.2. Effectiveness of Core Exit Temperature Measurement in Accident Management

An experiment performed in the OECD ROSA project demonstrated that the response of the core exit temperatures (CETs) was too slow and too late to be used as a reliable indicator for initiation of accident management procedures. The safety concern is that the CETs are used for initiation of emergency operating procedures and severe accident management in many countries. A wide range of CET values are being used as criteria for various accident management measures, indicating concerns about the uncertainties of the actual response.

A task group has been set up within GAMA to assess the technical basis for the use of CETs in accident management. The group should prepare a status report, covering the following items:

- Collection and review of the design basis of CET application for AM procedures in different countries
- Review of pertinent experimental results focusing on delay times between CET and core temperature rise.
- Conclusions and recommendations for further work.

5. CONTAINMENT

The CSNI activities regarding containment thermal-hydraulics have been presented in Session VII. Current activities include finalisation of the Containment Code Validation Matrix report, which defines a basic set of available experiments for the full range of ex-vessel phenomena expected in the course of light water reactor severe accidents. The report will be a supplement to former code validation matrices: 'Separate Effects Test Matrix for Thermal Hydraulic Code Validation, OECD/GD(94)82, September 1993' and 'In-Vessel Core Degradation Code Validation Matrix, OECD/GD (96)14, 1996'.

Another short-term activity proposed in GAMA is the development of best-estimate guidelines for lumped parameter analysis of containment TH. The idea is to help especially the newcomers in the field to produce high quality analysis and reduce the time spent to acquire experience. The selection of models, user-selected or user-defined parameters, nodalization of the containment, simulation of plant hardware, (e.g. igniters, recombiners, coolers, etc.) requires in-depth insight, therefore code developers and experienced code users will be involved.

5.1. OECD Projects

In the frame of the OECD SETH project a large number of experiments were carried out in the PANDA facility, examining three-dimensional mixing and stratification phenomena with steam and air or steam and helium as working fluids. PANDA consists of two interconnected, large cylindrical vessels that allows investigation of multidimensional effects relevant to power reactor containments at scales approaching those of actual containment buildings. The major achievements of the project – finished in 2006 – can be summarized as follows:

- Experimental investigation of the basic flow conditions controlling the mixing and inter-compartment transport: generation of high spatial and temporal resolution database
- Characterization of basic flow structures (plumes, jets), steam-air-helium mixing, propagation of stratification fronts and inter-compartment transport
- Code assessment and validation efforts as well as workshops focusing on the analysis of selected SETH PANDA Tests.

The SETH-2 project was started last year to complement SETH results with the aim to

- Improve understanding of mechanisms for gas stratification break-up by

  - Condensation (wall, containment cooler and spray)
  - Natural convection flows
  - Jets, negatively buoyant plumes,
  - Heat sources
  - Heat sinks
- Perform large scale containment tests:
  - Integral system tests with sophisticated and detailed measurements addressing LWR passive containment cooling system performance
  - Effect of sudden opening of rupture disks on gas mixing and stratification.

The tests will be performed in the PANDA and the MISTRA facility.

The THAI project – started in 2007 – also incorporates thermal-hydraulic tests focusing on the following questions:

- Demonstration of transferability of He test data to H2 distribution problems
- Better understanding of processes leading to formation and break up of a light gas stratification
- Confirmation/improvement of empirical PAR correlations for system codes
- Data generation on vertical combustion for safety assessment and code improvements
- Data generation for validation of coupled distribution/combustion models.

6. ADVANCED REACTOR ISSUES

There was no move in the advanced reactor subject in GAMA in the past few years, mainly due to the fact that activities were co-ordinated in different international channels, e.g. GenIV, EU. In the last GAMA meeting the ongoing and planned advanced reactor research activities at the USNRC were presented and potential international collaborative actions were discussed. The GAMA experience could be of benefit in adaptation and validation of computational tools for new reactor designs: ISP or similar benchmarking activities could be of benefit to many countries. Sufficient test material should be available for such activities. In this respect GAMA decided to evaluate research facility capabilities for gas-cooled and sodium reactors. In a first step the USNRC will provide feedback on this issue, after which a small group of GAMA members will make a proposal for such activity to be discussed in the next GAMA meeting in September 2008.

7. CONCLUSIONS

Activities related to primary system thermal-hydraulics have been decreasing in the past years, although a number of specific issues, e.g. boron dilution, heat transfer peculiarities in steam generators, water hammer problems etc. have been dealt with in the PKL and ROSA OECD projects. Most of the code validation work has moved to these projects as well that, however, means exclusion of the majority of the member states from the important output of these activities, since results of the OECD projects are non-public for a given period of time. This is why all OECD projects were requested to consider the possibility of releasing one of their tests for the purpose of an ISP.

The increasing use of CFD tools to solve specific problems raise new questions with respect to the requirements for these tools in nuclear reactor safety applications. The valuable efforts of the three Task Groups of GAMA have resulted in considerable progress: the Best Practice Guidelines are of great help in producing high quality CFD results and the review of the assessment base of different NRS issues gives the basis for further activity in the field. These actions will allow the validation and reliable application of CFD tools for a large number of single-phase flow NRS applications. For two-phase flows, several years will certainly be needed before quantitative NRS applications. Nevertheless, as those applications seem promising and will have significant consequences on safety evaluation, organisation of workshops, like CFD4NRS and XCFD4NRS are important steps to enhance international co-operation and understanding.

Significant progress was achieved in the field of Best Estimate methods including evaluation of uncertainties and the currently running BEMUSE project made an important contribution in this respect. Further activities will need to focus on identification of the uncertain parameters and more rigorous definition of the related distributions. Extension of the evaluation of uncertainties to coupled phenomena, e.g. to coupled 3D neutronics / thermal hydraulics, is a challenging task, which has recently been assumed by NSC.

Tasks addressing regulatory concerns, e.g. sump strainer clogging or core exit temperature measurement effectiveness for accident management are of high priority that necessitate the formation of special task groups and/or close collaboration of GAMA with other CSNI working groups.

New plants or future reactors might need advanced methods in order to solve specific issues and they may also highlight new problems. A periodic review of features, which can challenge existing models or tools should be performed in order to anticipate, as far as possible, new developments necessary in safety analysis.

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