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Review

The working group on the analysis and management of accidents (WGAMA): A historical review of major contributions

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ABSTRACT

The Working Group on the Analysis and Management of Accidents (WGAMA) was created on December 31st' 1999 to assess and strengthen the technical basis needed for the prevention, mitigation and management of potential accidents in NPP and to facilitate international convergence on safety issues and accident management analyses and strategies. WGAMA addresses reactor coolant system thermal-hydraulics, in-vessel behaviour of degraded cores and in-vessel protection, containment behaviour and containment protection, and fission product (FP) release, transport, deposition and retention, for both current and advanced reactors. As a result, WGAMA contributions in thermal-hydraulics, computational fluid-dynamics (CFD) and severe accidents along the first two decades of the 21st century have been outstanding and are summarized in this paper. Beyond any doubt, the Fukushima-Daiichi accident heavily impacted WGAMA activities and the substantial outcomes produced in the accident aftermath are neatly identified in the paper. Beyond specific events, most importantly, around 50 technical reports have become reference material in the different fields covered by the group and they are gathered altogether in the reference section of the paper; a common outstanding feature in most of these reports is the recommendations included for further research, some of which have eventually given rise to some of the projects conducted or underway within the OECD framework. Far from declining, ongoing WGAMA activities are numerous and a number of them is already planned to be launched in the near future; a short mention to them is also included in this paper.

1. Introduction

The Working Group on the Analysis and Management of Accidents (WGAMA) was created on December 31st' 1999, by combining two previous Primary Working Groups of PWG-2 and PWG-4, each of them were primarily in charge of in-vessel and ex-vessel thermal-hydraulics and severe accident phenomena in nuclear power plants (NPP). It is one of the nine working groups (WGs) of the Committee for the Safety of Nuclear Installations (CSNI), which is responsible for the promotion of Nuclear Safety among the 33 member states of the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA). Fig. 1 shows the CSNI WG structure and OECD/NEA joint projects; the WGAMA is highlighted.

The overall WGAMA objectives are to assess and strengthen the technical basis needed for the prevention, mitigation and management of potential accidents in NPP and to facilitate international convergence on safety issues and accident management analyses and strategies. In order to fulfil these objectives, the WGAMA exchanges technical experience and information relevant for resolving current or emerging safety issues, promotes the development of phenomena-based models and

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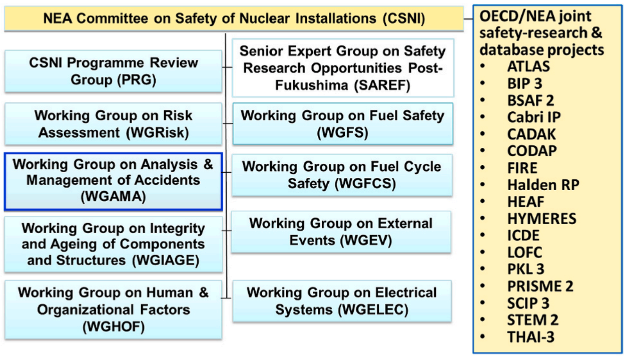


Fig. 1. Working Groups and Project under the NEA/CSNI framework.

codes used for the safety analysis, assesses the state of knowledge in areas relevant for the accident analysis and, where needed, identify and fosters research activities aimed to improve such understanding, while supporting the maintenance of expertise and infrastructure in nuclear safety research. The intention is always to make significant contributions to the regulatory decision-making on prevention and management of accidents, reactor safety capacity building, continuously improving the state of knowledge and knowledge management.

The specific scope of the WGAMA extends over a number of fields associated with both current and advanced reactors: reactor coolant system thermal-hydraulics; in-vessel behaviour of degraded cores and in-vessel protection; containment behaviour and containment protection; and fission product (FP) release, transport, deposition and retention. Such a vast scope requires involving large resources, which in numbers translates into more than 100 registered WG members (the largest CSNI WG), over 200 scientists and engineers involved in the running of activities and in excess of 25 technical reports published in the last 5 years, most of which have already been heavily referenced via web referencing and downloaded from the NEA/CSNI document directory (https://www.oecd-nea.org/nsd/docs/indexcsni.html).

The present paper gives an introduction of WGAMA by briefly reviewing the main technical contributions completed since its inception in the fields of thermal-hydraulics, computational fluid-dynamics (CFD) and severe accidents. The WGAMA history has been split into three time periods: decade preceding the Fukushima-Daiichi accident (2000–2010), in which WG's activities and dynamics got consolidated; the next five years (2011–2015), which were heavily influenced by the Fukushima-Daiichi accident; and the present times (2016–2019), in which the group is looking ahead based on the recent and past experience since its inception. Given the limited extension of this article, only the most significant contributions are here shortly described, but a more thorough overview may be done by going through the NEA/CSNI document directory (https://www.oecd-nea.org/nsd/docs/indexcsni.html).

2. The pre-Fukushima Daiichi "decade" (2000–2010)

2.1. Thermal-hydraulics

The safety of NPPs is demonstrated and assessed through deterministic safety analysis with conservative inputs, to confirm adequate margin to an acceptance criterion well below the failure point, namely of the nuclear fuel. The use of best-estimate (BE) computer codes, combined with conservative or realistic input data, has also been gaining acceptance at the latter part of the 20th century. The code scaling capability and uncertainty methods developed in the same period evolved simultaneously, as the benefits of more realistic safety analysis became increasingly clear. The evaluation of the uncertainties in calculated results is an integral part of BE methods (known as BEPU methodologies). Therefore, sensitivity and uncertainty analysis have taken the centre stage of nuclear reactor system design and safety.

In 2000, it was proposed that WGAMA develops an action plan on Best Estimate calculations and uncertainty analysis (NEA, 2002a). A workshop was held to review BE methods already in use for safety analysis together with an attempt to identify weaknesses such as difficulties in implementation and justification of basic assumptions. In addition, methods not being used in safety analysis at that time were also reviewed. One of the major recommendations was to launch computational benchmark exercises using separate effect tests and integrated effect experiments.

A primary contribution of WGAMA has been the documentation of the code validation matrix for reactor cooling system and containment thermal-hydraulics. Most of the data archived in the NEA Data Bank (separate-effect and integral tests) come from NEA Joint Projects that, after a confidentiality period typically 3 years, are made available to NEA members. Several sets of international experimental data (a sample is shown in Table 1) have been made available for performing code evaluations and assessments, occasionally in the form of International Standard Problems (ISP) framed under WGAMA.

A sizable work program was established under the acronym of BEMUSE (Best Estimate Methods Uncertainty and Sensitivity Evaluation) to evaluate the practicability, quality and reliability of BE methods (NEA, 2006). In addition, an evaluation of uncertainties for nuclear reactor safety applications was performed to promote the use of these methods by the regulatory bodies and nuclear industry. The activity was divided into five phases. As test cases, the BEPU analyses of the LOFT[1] L2-5 test and of a Large Break Loss-Of-Coolant Accident (LBLOCA) were proposed. This challenging activity concluded that the practicality and/or validity of these BE codes was not adequately established to support general use to be accepted by industry and safety authorities (NEA, 2006). A systematic qualitative and quantitative accuracy evaluation of simulation results was performed, along with sensitivity analyses, and the lessons learned were used as guidance for deriving uncertainty bands in subsequent phases of BEMUSE. This initial BEMUSE activity attracted 14 participants with 6 thermal-hydraulic codes.

The next step within BEMUSE was to simulate a cold leg double-ended guillotine LBLOCA with an assumption of the high-pressure injection system unavailable in a 4-loop 1040 MWe Westinghouse PWR (Zion NPP). The scenario was successfully simulated and the main parameters predicted with credible consistency. The predicted peak cladding temperatures (PCT) were quite close to one another (Fig. 2), although the temporal variation of PCT and the timing of complete core rewet were inconsistent. The major differences among calculated results came from reflooding behaviour and timing and they were attributed to code effects. The exercise helped to understand the nuances of user effect and the need to differentiate user effect and code effect. In addition, participants (13 organizations; 6 thermal-hydraulic codes) were able to identify the parameters that strongly affect PCT and reflood.

The final stage of the BEMUSE activity was to complete an uncertainty analysis of a LBLOCA analysis, using the Zion exercise (NEA, 2008) as a reference calculation to obtain uncertainty bands for a number of variables (i.e., PCT; upper plenum pressure; time of complete core quenching; etc.). Two methods were applied to obtain uncertainty bands: the propagation of input uncertainty, and the propagation of output accuracy. The former method was used by 12 participants whereas just 2 participants used the latter. A main result of the activity was that all participants managed to obtain uncertainty bands with reasonable width (NEA, 2009c). The overall results from the study were a step forward to consolidating the different methods, although familiarity with the technique was recommended to participants using the probabilistic approach.

The conclusions and recommendations mentioned in the previous

[1] Loss-Of-Fluid Test facility located in Idaho National Laboratory, USA.

Table 1
CSNI thermal-hydraulic code validation matrix.

Facility	Facility Location	Facility Description	Phenomena Covered
BETHSY	Nuclear Research Center, Grenoble, France	Scaled down model of a 3 loop, 900 MWe, FRAMATOME PWR	Critical 2-phase flow, loop seal clearing, heat-transfer during boil-off or accumulator injection, and primary side refilling by Low Pressure Injection System
DOEL2	Belgium, TRACTEBEL	Westinghouse, 2 loop PWR, 392 MWe	Steam Generator Tube Rupture (SGTR) incident on full scale facility
FIST	USA, General Electric Co., San Jose, California.	BWR/6–218 standard plant scaled to 1/624 of the system components.	Channel and bypass axial flow and void distribution and core heat transfer including DNB, dryout, RNB, surface to surface radiation
FIX-II	Sweden, Studsvik Energiteknik	Based on Oskarshamn 2 BWR	Mass flow and power transients following a total loss of power to the recirculation pumps. Determined initial power limit to give dryout and peak cladding temperature.
LEIBSTADT	Switzerland, Leibstadt Nuclear Power Station	BWR, 3012 MWt, 942 MWe	The liquid level to demonstrate the capability of the Reactor Core Isolation Cooling (RCIC) system to maintain the reactor water level.
LOBI	Italy, EC Joint Research Centre, Ispra	1/700 scale model of 4 loop PWR with 5.3 MW electric-heated rods	Natural circulation in single and two-phase flow; asymmetric loop behaviour; phase separation; ECC mixing and condensation; heat transfer.
LOFT	USA, Idaho National Engineering Laboratory	Loss-of-Fluid Test (LOFT) facility is 50 MWt PWR extensively instrumented.	Phase separation; ECC bypass and penetration; core heat transfer; quench front propagation, effect of primary coolant pump restart on core cooling when primary coolant system highly voided.
MARVIKEN	Sweden, Studsvik Eco & Safety AB	Marviken Power Station, unfinished direct-cycle BHWR	Full Scale Containment Blowdown experiments
OTIS	United States, Babcock and Wilcox at Alliance, Ohio	Once Through Integral Test Facility (OTIS) is 180 kW facility representing 10% scaled power of a 2584 MWt PWR.	Natural circulation in 1-phase flow, boiler condenser mode, Leak flow, Heat transfer in covered core, non- condensable gas effects, intermittent two-phase natural circulation, and natural circulation in core, vent valve, downcomer, superheating in secondary side
PACTEL	Finland, VTT Energy	Russian design VVER-440 (PWR) scaled to 1/305 vol and power	Natural convection circulation in a VVER plant
PIPER	Italy, Dell'Energia Nucleare e delle Energie Alternative (ENEA)	General Electric BWR-6 plant scaled to 1/2200 vol and 1/1 height.	Energy and mass balance of the loop using extended instrumentation.
PKL	Germany, Gesellschaft fuer Reaktorsicherheit (GRS) mbH, Munich	Simulates West German 1300 PWR at scale 1:134. Elevations and locations are essentially full scale.	There are numerous measuring points for determination of thermohydraulic phenomena in the pressure vessel and in the loops.
ROSA-III	Japan, Japan Atomic Energy Research Institute	Full-pressure half-height 1/124 scale facility for BWR thermal-hydraulic response during LOCA with four electrically heated fuel bundles	Core thermal hydraulics; parallel channel effects and instabilities; void collapse and temperature distribution during pressurization; critical power ratio.
ROSA-IV/ LSTF	Japan, Japan Atomic Energy Research Institute	Full-pressure full-height 1/48 scale facility for PWR thermal-hydraulic response during small break LOCA or operational transient.	Visual observation of the flow in the primary loops using high-pressure video probes located at the inlet and outlet legs of the two SGs.
SEMISCALE	USA, Idaho National Engineering Laboratory	Scaled to 1/1705.5 of a reference four-loop PWR with an electrically-heated, 25-rod PWR core simulator.	Natural circulation in single and two-phase flow; core thermalhydraulics; single and two-phase pump behaviour.
SPES	Italy, Dell'Energia Nucleare e delle Energie Alternative (ENEA)	SPES integral test facility is 3-loop scale model of PWR with 1:427 power-scaling ratio.	At various locations the following were examined: void fraction, temperature, pressure drop, mass flow, power, pressure, downcomer level, coolant mass, and density.

paragraphs and some others were synthesized in a summary report, compiling the main insights stemming out of this long activity (NEA, 2011a). Although there was consensus on the maturity of BEPU application to accident analysis using system computer codes, there is a wide spectrum of diverging views in other new areas like handling of uncertainties in coupled codes, extension of BEPU methods to PSA environment, etc. The key technical and regulatory issues requiring further work are management of residuals, implications on the definition of Technical Specifications, etc. It was recommended to pursue these efforts to achieve a consensus on the methods of quantification of input epistemic uncertainties and their treatment. A follow-up activity, called the PREMIUM project, was launched to address the issue.

Improving the understanding of Rapid Boron-Dilution (RBD) transients was studied in an International Standard Problem (ISP) No. 43 exercise (NEA, 2000). Boron being a neutron absorber is used for reactivity control in PWRs and for the same reason is also injected into water during emergencies to decrease core reactivity. Sudden localized deboration of water leading to reactivity excursion is analyzed as part of safety analyses. The ISP initiated a test program to study the formation of deborated water slug in a stagnant section of the primary system.

Two test series, A and B, were completed in the facility shown in Fig. 3 to characterize RBD transients in order to get qualified data required for a benchmark study. The efficacy of codes to model major geometric and operational features, encountered in a prototype reactor, was assessed. The code predictions and experimental data were compared only in the downcomer region. The first test series was a simple special effect type exercise while the last test series involved all of the features of the integral test facility. In the tests A, the liquid front was injected from an external tank at the bottom of the steam generator, by the primary coolant pump in one cold leg, while the other cold legs remained isolated. The primary system was then isolated, and the water slug was set in motion by the pump. The slug followed a closed path through the cold leg, the downcomer, the lower plenum, the core, and returned to its initial position through the hot leg of the loop. In test series B, all cold legs participated and bypass was made possible in all non-injection cold legs.

The test series provided significant insights into code capabilities to capture thermal-hydraulic phenomena important to RBD transient. The first test series showed that buoyancy has an impact on the transient and therefore temperature dependent properties are essential for accurate simulation of the phenomena, because in the real boron dilution transient, the temperature differences between the slug and the primary coolant into which it is injected can be substantial. The ISP activity concluded that despite very high Reynolds numbers expected in the prototype pump startup, an examination of downcomer Froude number must be made before running plant simulations.

The international nature of activities in WGAMA prompted the development of an experimental test facility matrix for the validation of best estimate thermal-hydraulic computer codes applied for the analysis of VVER reactor primary systems in accident and transient conditions (NEA, 2001b). The activity identified phenomena relevant to VVER reactor primary and secondary systems during LOCAs and transients and compared the phenomena of VVER reactor systems with LWR reactor systems to highlight similarities. Following this, the relevant

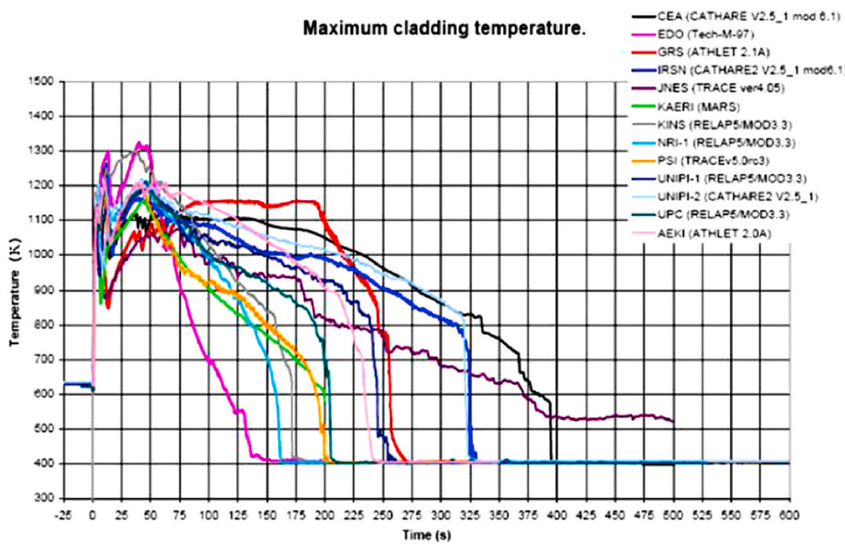


Fig. 2. A comparison of calculated peak cladding temperatures in BEMUSE (Zion exercise).

phenomena were identified and described in detail to identify test facilities and experiments that supplemented the CSNI CCVMs developed in 1987 for the LWR community.

A review of Core Exit Temperature (CET) thermocouple effectiveness for accident management was prompted following OECD/NEA ROSA project with LSTF test[2] which simulated a vessel head small break loss-of-coolant accident (SBLOCA) under the assumption of total failure of the high pressure injection system (NEA, 2010). The test had to be terminated prematurely to avoid excessive overheating of the core. During the test, the core uncovery had started well before CET thermocouple indicated superheating. The rate of temperature increase in the core was higher than shown by the CET. The results suggested that the response of the CET thermocouples could be inadequate to initiate the relevant accident management actions. Examples from LOFT, PKL, and LSTF tests seemed to corroborate this observation. During the CET study, extensive review of different sources and experiments were completed. The study concluded that CET reading is generally used by NPPs during: (1) Emergency operations in order to prevent accidents, (2) the transition from emergency operations to severe accident management guidelines, and (3) the transition from severe accident management guidelines to mitigation accident management. There were models available to establish the link between CET measurements and the maximum cladding temperature, however, they had doubtful validation basis. The study therefore concluded that the delayed CET response was a concern. The activity recommended: (a) Further studies to assess heat transfer models affecting CET behavior, (b) development of best practice guidelines for the nodalisation of uncovered core, (c) additional comparison with test results to understand 3-D effects, and (d) applying CFD methods, if 3-D effects are found to be dominant.

Within the area of thermal-hydraulics and starting around the year 2002, Transfer of Competence, Knowledge and Experience gained Through CSNI activities in the field of thermal-hydraulics, THICKET, was organized. Devoted one-week seminars were conducted in 2004 (Paris), 2008 (Pisa), 2012 (Paris) and 2016 (Budapest). The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (OECD-NEA) approved the next Seminar planned in 2020 (Toronto). Lecturers have always been WGAMA senior experts who contributed to the thermal-hydraulics activities in decades before the Seminar and participants (in each case in the range 20–30) were junior experts who had the idea to pursue CSNI activities in their future. THICKET definitely encompasses all the three periods considered in the present paper and will not be discussed further.

2.2. CFD applications in nuclear safety analyses

The use of 1-D network to model flows in reactor cooling system and containment is known to be too simplified to capture the complexity that governs accident scenarios in the upper and lower plena, downcomer and reactor core and natural circulation, mixing and stratification in containment, where the flow is 3-D. However, the application of CFD codes in nuclear safety took longer than in other industries, at least partially due to scarcity of high-fidelity data necessary for validating analytical tools in the field. At the beginning of the 21st century, an action plan to apply CFDs to nuclear safety was defined (NEA, 2002b). Since then, computational power, computational methods and instrumentation technology have advanced substantially.

Major highlights in the WGAMA action plan were: the identification of nuclear safety scenarios in which CFD application might help in understanding; the need for developing reference guidelines for nuclear reactor safety (NRS) application (NEA, 2007c, 2014a); and the convenience of organizing workshops of CFD4NRS, computational benchmark

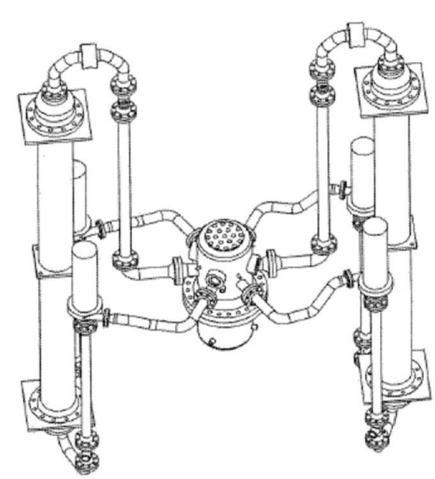


Fig. 3. The schematic of the University of Maryland 2x4 B&W Loop Facility where ISP-43 tests were completed.

[2] ROSA - Rig of Safety Assessment in Japan Atomic Energy Agency; LSTF - Large Scale Test Facility volumetrically scaled to 1/48 of Westinghouse type 3423 MW_{th} four loop Pressurized Water Reactor.

exercises, and ISPs.

Three workshops were held in between 2006 and 2010 (NEA, 2007b; NEA, 2009d; NEA, 2011b). The first one highlighted specific areas of validation (i.e., buoyancy, stratification, turbulence mixing; etc.) and it was highly recommended to hold specialized workshops to keep the momentum, monitor progress and enhance knowledge transfer. The second one emphasized the importance of establishing a close link between experimentalists and analysts, so that model and codes could be more efficiently developed and the experiments could become truly useful for validation; in particular, concepts were proposed for CFD-grade data with measurement uncertainty estimates for validating CFD models and for applying uncertainty evaluation methods to CFD codes. The third workshop assessed the status of CFD code capabilities and validation, exchanged experiences in CFD code applications and monitored progress.

As an additional outcome of the third workshop, an exercise on thermal fatigue in a T-Junction, where hot and cold streams of liquids merge and mix, was organized based on an *ad-hoc* experiment (Fig. 4).

Overall, the T-junction benchmark was successful (Smith et al., 2011). The exercise complemented activities in other areas and helped in understanding the origins of thermal fatigue in this geometry. Different codes, different modelling approaches, and different numbers of control volumes were adopted by the various participants, and useful information was forthcoming from those who used the same code. Fig. 5 compares non-dimensional measured and calculated T* values for a select number of submissions denoted as S21, S16, S20 and S2. A simple heat balance calculation showed that $T^* = 0.38$ for perfect mixing of the streams. The asymptotic dissipation in the left-hand-side graph shows this trend very well in the experiment and calculations, where turbulent mixing gradually overcame the initial flow stratification.

2.3. Severe accidents

The first decade of the century resulted in a substantial enlargement of the NEA database on severe accidents through joint projects like: SERENA on steam explosions; MASCA on in-vessel corium behavior; THAI on hydrogen (H_2) and FP behavior in containment; BIP on in-containment iodine behavior; and SETH2 on H_2 stratification break-up in containment. No specific references are included here because, despite their close link to WGAMA, they are not WGAMA activities but the NEA Joint Projects that provide final synthesis reports on the CSNI website (https://www.oecd-nea.org/nsd/docs/indexcsni.html).

A good number of international standard problems (ISP) were launched on different severe accident issues: iodine behavior (NEA, 2001a; NEA, 2004a); hydrogen generation during core reflood (NEA, 2002c); aerosol depletion in wet containment atmospheres (NEA, 2003); containment thermal-hydraulics (NEA, 2007d); and H_2 combustion (NEA, 2011c). Additionally a code benchmarking was set around an alternative scenario of the Three Mile Island accident (NEA, 2009a) as a way to assess the code capabilities at the time. Nonetheless, the most interesting calculation exercise conducted due to data representativeness and comprehensiveness of the scenario was the ISP-46 on the PHEBUS-FPT1 experiment (NEA, 2004b).

The ISP-46 was participated by more than 30 teams who submitted 47 base case calculations and 21 additional best estimate calculations with 15 different codes. Major insights were gained from the data-estimate comparisons and from the code-to-code comparisons. Whenever key parameters were rightly chosen, the thermal-hydraulic evolution and the H_2 production during the core degradation showed good agreement and the final state of the core was reasonably predicted. Fig. 6 shows code predictions along with data (black dash line) for core temperature and cumulative H_2 generation. Even though volatile fission products release were well predicted, large discrepancies existed when coming to semi-volatile and low-volatile fission products, which indicated some additional modeling needed. But the major difficulties in getting good estimates concerned iodine behavior in the PHEBUS containment, where there was a large scatter when iodine gas (particularly organic iodide) concentrations were estimated. Fig. 7 plots the in-containment organic iodine predicted and measured (red dots) during the FPT1 experiment. At the time organic iodine production was not well understood and, as a consequence, existing models were rather premature and based on a scarce database; such discrepancies were the seeds of OECD projects proposed in the coming years, like BIP and STEM.

In addition, two state of the art reports (SOARs) were produced and have ever since inspired most of the source term research conducted. The SOAR on iodine chemistry (NEA, 2007a) thoroughly collected all the relevant information available in the multiple aspects of iodine chemistry: iodine release from fuel; iodine transport through the circuit; iodine behavior in containment (including liquid and gaseous chemistry); and, finally, the iodine source term to environment. The importance of the insights gained from the PHEBUS project, along with other ongoing bench-scale test projects at the time (i.e., EPICUR, PARIS and CHIP), was clearly stated too. In addition, some gaps, like the complexity of production/destruction of organic iodides, were identified and some recommendations were made. Some of those recommendations were later addressed by experimental projects, like BIP and STEM.

The SOAR on nuclear aerosols (NEA, 2009b) was the third one sponsored by CSNI on the matter. It resulted from a recommendation made in an OECD workshop held in 1998. It contains background chapters summarizing the fundamentals of aerosol physics with a long list of references, but the main body of the report deals with aerosol models and their validation and gives examples of their application in plant analyses. It concluded that there still existed a number of items for which additional work was required, some of which were: mechanical

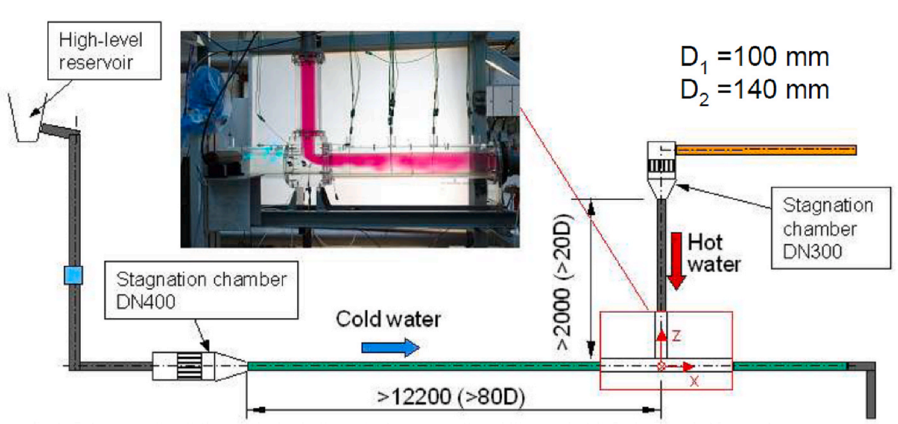
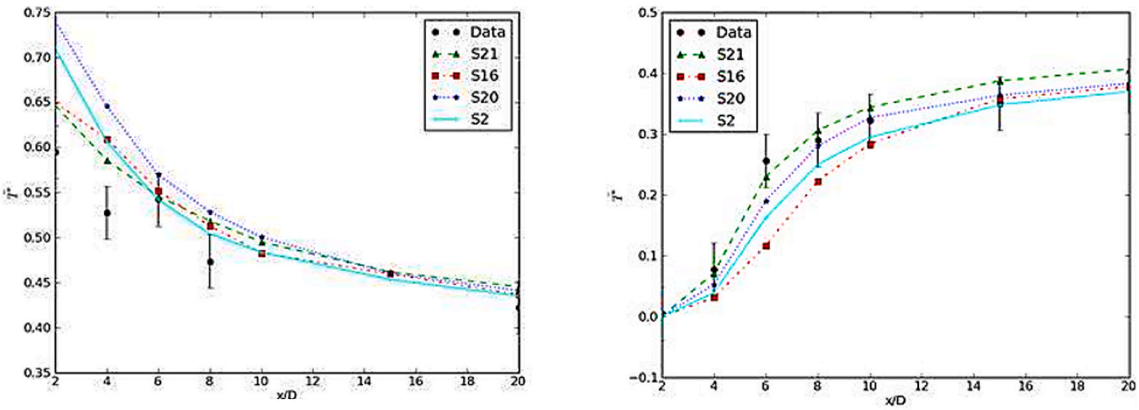


Fig. 4. A schematic and a photographic view of Vattenfall T-Junction test apparatus.



(0°) (180°)

Fig. 5. Time-averaged non-dimensional temperature at the downstream pipe.

resuspension, particle break-up in highly turbulent flows, revaporisation, reentrainment, particles chemistry and interaction with hydrogen burns and some others. However, the status of aerosol codes and experimental data-bases was recognized to have improved substantively since the publication of previous SOARs. In this regard, the contributions from the KAEVER, VANAM, ARTIST and THAI tests were recognized and the add-on of the integral PHEBUS tests was highly valued. Nonetheless, the report highlighted that there remained a need to harmonise code user practices with respect to plant analyses to reduce divergence in results and suggested the idea of producing "best practice" guidelines for the major codes.

3. The Fukushima DAIICHI years (2011–2016)

The five year period from 2011 to 2016 was soon heavily conditioned by the Fukushima-Daiichi accident that happened on March 11th, 2011. This clearly marked the activities that WGAMA faced with; nevertheless, the momentum in other areas was also kept as briefly reported below.

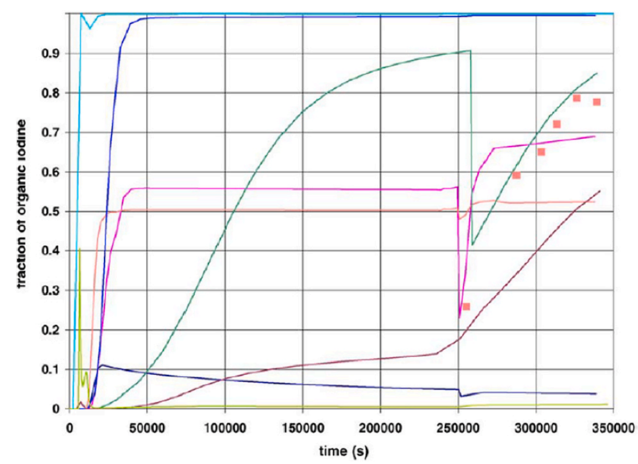
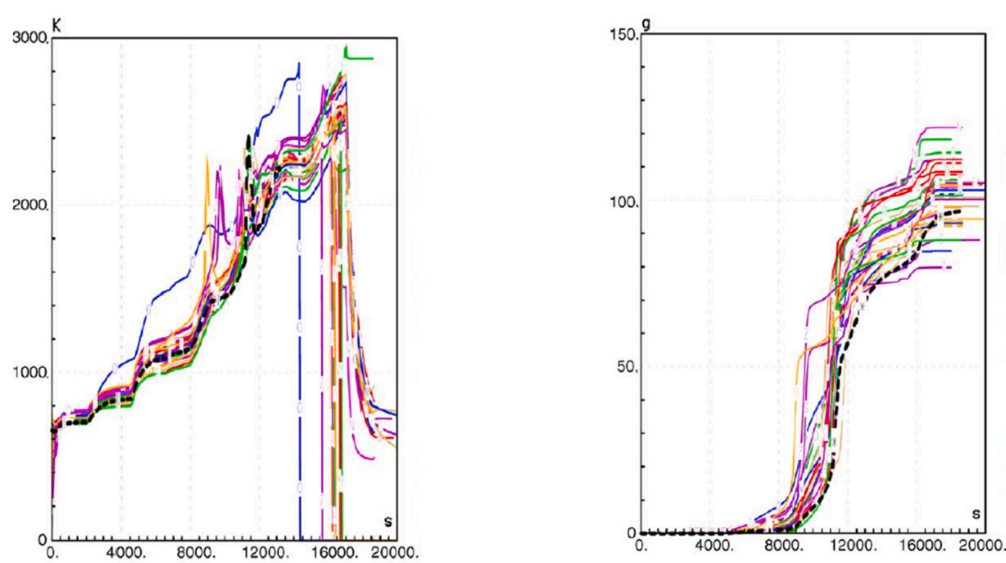


Fig. 7. Organic iodine concentration in PHEBUS FPT1 containment.



Average Temperature at the outer ring H$_2$ cumulative production

Fig. 6. Variables describing core status during the PHEBUS FPT1 degradation phase.

3.1. Thermal-hydraulics

The importance of scaling in nuclear technology is recognized as a major issue by thermal-hydraulic experts. The origin of the scaling issue is entrenched in the difficulty to demonstrate that a model behaves like the prototype. A variety of approaches have been used to address the scaling issue, including: non-dimensional analysis of mass-, energy- and momentum-balance equations; the derivation and application of scaling factors, including the hierarchy of relative importance; performance of experiments at different scales; and, finally, running system thermal-hydraulic computer codes.

A WGAMA activity on scaling was started to provide expert insights into the controversial scaling issue and a comprehensive State-of-the-Art Report (SOAR) was produced (NEA, 2016a). The document discusses the key areas and approaches to scaling, which was defined as the process of converting any parameters of the nuclear power plant, to those either in experiments or in the results of numerical codes, to reproduce the dominant prototypic phenomena in the model. Scaling distortions may result from assumptions and simplifications in scaling methods, from technological limitations in constructing and operating test facilities, and from limitations of computer code scalability. By using the Buckingham Pi theorem, or by writing conservation laws in non-dimensional form on a selected global or local control volume, a list of non-dimensional groups is generated which define similarity conditions. However, all the non-dimensional groups cannot be matched simultaneously in the design of reduced-scale test facilities, resulting in some scaling distortion.

Eight different scaling methods with major characteristics, merits, limitations, and application areas have been discussed. These methods are: linear scaling; power-to-volume scaling; three-level scaling; hierarchical two-tiered scaling; power-to-mass scaling; modified linear scaling; fractional scaling analysis; and dynamical system scaling. Depending on the experimental objectives, as well as budget and facility building constraints, these approaches use scaling height (volume), time and/or pressure.

Another key aspect of scaling was the applicability of computer code calculations to reactor systems. When code applicability for an analysis has been determined, uncertainty in the predicted safety parameters has to be determined and the scaling distortion has to be considered in the uncertainty evaluation process. It is likely that scaling distortion can become large, and it is difficult to determine the acceptability criteria for distortion in an experiment. The effects of such distortions require a method that can evaluate the accumulated distortion of a process as a function of time. Something worth highlighting is that predictions accuracy of system thermal-hydraulic computer codes when applied to differently-scaled experiments, may not depend upon the scale of the experiments used in validating the code.

The scaling technique used to design the test facility is key to understand the validity of experimental data. Scaling parameters for a local phenomenon can be derived by applying a dimensional analysis (empirical approach), or dimensionless governing equations (a mechanistic approach). An empirical approach uses correlations and models to derive similarity parameters, or to estimate distortions due to scaling, such as the criterion for flow regime transition based on the Froude number. The approach of the dimensionless governing equation is to simplify the governing equations for both the prototype and model by making assumptions and evaluating the various terms; the similarity criteria can be obtained by comparing the non-dimensional terms in the equations.

To preserve kinematic and dynamic similarities between the prototype and scaled test facility, a scaling method at system level is necessary. Most scaling laws are derived from the non-dimensional governing equations. For integrated test facilities, another level of scaling needs to be used by preserving the important local phenomena and reducing scaling distortions. The important phenomena and processes can be identified from the phenomena identification and ranking table (PIRT).

How well system analysis codes are able to calculate a 50% Direct Vessel Injection (DVI) line break was the benchmark objective in the OECD/NEA ISP-50 exercise (NEA, 2012). A DVI line break integral effect test was performed in the Advanced Thermal-hydraulic test Loop for Accident Simulation (ATLAS) operated by Korea Atomic Energy Research Institute (KAERI). The ISP-50 exercise was completed in two consecutive phases: a blind phase and an open phase. The main objectives of the blind phase were to assess capability of the current leading safety analysis codes in reproducing the thermal-hydraulic phenomena relevant to a DVI line break scenario and to investigate the magnitude of "user effects". The ISP-50 was the first-ever international cooperative study that focused on the DVI line break LOCA as well as the direct vessel injection of ECC water. The open phase of the exercise assessed the prediction capability of the multi-dimensional behaviour observed in the ISP-50 test, especially in the annulus downcomer region and examining how much the "user effects" can be reduced for given integral effect data. In addition, limitation of the current safety analysis codes was investigated with areas where further code improvement can be made. In general, the ISP-50 gave a wide ranging and very valuable outlook of the actual status of code performance. Various codes were tested against the same test data while evaluating the output differences in the same code due to the differences in users. These user effects (i.e., the effect that user's decisions concerning multiple aspects of the modeling, such as choices made in input deck options, nodalization schemes or even misinterpretation of input cards, do have on the code results) were observed in the blind phase, confirming that these effects are still one of the major issues for system thermalhydraulic code applications.

3.2. CFD applications in nuclear safety analyses

A benchmark activity was organized to test the ability of state-of-the-art CFD codes to predict important turbulence parameters downstream of a generic spacer grid design in a rod-bundle geometry (NEA, 2013b). The data required for the benchmark activity was obtained from the MATiS-H (Measurement and Analysis of Turbulent Mixing in Sub-channels—Horizontal) experimental facility. The sketch of the test section is shown in Fig. 8. Experiments were conducted for two spacer grid designs. The Reynolds number, based on the hydraulic diameter within the rod bundle, was ~50000, corresponding to an axial bulk velocity of ~1.50 m/s within the bundle region. Detailed measurements of the velocity field were taken using a 2-D Laser Doppler Anemometer (LDA) system at four downstream locations from the spacer grid. Flow conditions (mean and fluctuating velocities) were measured upstream of the spacer grid, to provide suitable inlet boundary conditions for the CFD simulations.

A total of 25 participants submitted blind CFD calculation results; the majority (19) were obtained using three commercial CFD software packages: ANSYS CFX, FLUENT and STAR-CCM+. The remainder used the open-source CFD software OpenFoam (1) and various in-house CFD codes (5).

An overall good agreement with experimental data was achieved with a moderate number of mesh cells, and less sophisticated turbulence models, provided that care is taken to follow Best Practice Guidelines, and first-order space discretization is avoided. Generally, the scale-resolving turbulence models performed well, while the simple Reynolds Averaged Navier Stokes (RANS) models captured the mean velocity profiles quite well, but showed evidence of being over-diffusive for this application. In complete contrast to the previous T-junction benchmark, the Scale Adopted Simulation (SAS) turbulence model performed well in this exercise, but is still inferior to full Large Eddy Simulation (LES).

The development of the mean vorticity component in the axial direction (z) is a useful measurement that can be compared with calculations (Fig. 9. The integrated circulation along the sub-channel is shown in the figure for the two top-ranked simulations Sw02 and Sw07 (both full LES) and for Sw04, a middle-ranked simulation featuring the Baseline Reynolds Stress turbulence model. The circulation at $z = 0.5$ hydraulic diameter (D_H) and $z = 1.0\ D_H$ are under predicted by Sw02 and Sw07, possibly due to the integration by the vorticity close to the rod surfaces. By comparison, Sw04 shows good agreement with measured values. As the secondary vortices decay downstream, the circulations derived from the Sw02 and Sw07 results recover, and compare very well with experiment at $z = 10.0\ D_H$ (see Fig. 10).

As a result of the exercise, a more unified numerical approach to the use of CFD in spacer grid design has been established.

As said above, at the conclusion of BEMUSE (NEA, 2011a) activity, the importance of uncertainty quantification became quite obvious. A logical outcome of this realization was the activity, rightfully termed as the Post-BEMUSE Reflood Model Input Uncertainty Methods (PREMIUM). The task devoted itself to the various methods of quantification of uncertainty in physical models contained in thermal-hydraulic system codes, used in simulating specific thermal-hydraulic scenarios relevant to nuclear reactor safety. The uncertainties in the codes were quantified using a benchmark against "intermediate" separate effects tests. These separate effects tests involved only a few phenomenon and models and as such the dependency on expert judgement was eliminated in the quantification of uncertainties. The scenario selected for the benchmark assessed the models involved in the reflooding predictions of system thermal-hydraulic codes. A number of thermohydraulic codes were used in the benchmark, however, the results were very dependent on the uncertainty quantification method rather than on the codes used. The results of uncertainty quantification had a strong dependence on: (a) The set selected to track the responses, (b) the set of parameters selected for quantification, (c) he data selected for quantification, and (d) the models and their numerical implementation used in the quantification. The quantified uncertainties in PREMIUM benchmark had large variation and inconsistency among participants. The benchmark also found that there was a dire need for best practice guidelines on this topic. The participants, in the absence of a generalized guideline, took a number of decisions that varied based on experience and or procedures available to them. Once again, PREMIUM benchmark outcome showed a strong influence of user effect. One of the main lessons learned is that the calculated uncertainty, in an output parameter, is strongly dependent on the set of input parameters selected for uncertainty quantification. This means that quantified uncertainties are attributes of the total set of parameters, rather than intrinsic properties of individual parameters. Thus, the key message from the study is that the set of quantified parameters must include the parameters that have most influence on the output parameter; otherwise the resulting uncertainty may be completely misleading. As a conclusion, extrapolation of quantified uncertainties (i.e. application to forward calculations outside the range of validity) may lead to erroneous results.

A new test was carried out during February–March 2013 within a comprehensively-instrumented vessel of the PANDA integral containment facility, located at PSI in Switzerland to provide a CFD grade experimental data for another benchmark exercise on stratified helium (a simulant for hydrogen) layers (NEA, 2016c). The benchmark specifically examined the erosion of a helium-rich, stratified layer, occupying the upper reaches of a containment volume (see)Fig. 10). Helium rich layer at the top of the vessel was impinged upon by a buoyant, axially off centre, vertical helium jet to induce three-dimensional motions in the flow (Fig. 9). Temperature and gas concentration were measured at strategic locations throughout the volume and Particle Imaging Velocimetry (PIV) was used to measure the vertical and lateral velocities of the helium jet expansion and interaction with the ambient air at room temperature. The facility took extra care in providing measurement error and jet injection boundary conditions. During the test, a helium rich stratified layer was established at the top of the vessel dome. The jet buoyancy was established with air surrounding the area below the stratified layer and with increased helium concentration at injection, and with a slightly elevated temperature.

There were 49 registered participants, 19 of which submitted blind CFD results and presented the findings as invited papers at the joint CFD4NRS-5 Workshop held in Zurich during September 2014. Of the simulation results submitted, the majority (twelve) were obtained using the three major commercial CFD software packages: ANSYS CFX, FLUENT and STAR-CCM+. The rest were derived from the open-source software OpenFOAM (one), various in-house CFD codes (four) and from the dedicated containment modelling code GOTHIC (two). These same participants were offered full access to the test data as an incentive for their participation. The benchmark concluded that containment modelling is still remaining to be a significant challenge for CFD codes and for those who use them, even in the absence of complex physical phenomena such as condensation. A comparison of measured helium concentration at one of the uppermost location is shown in Fig. 11. The best comparisons with experimental measurements were obtained from those with previous experience in the simulation of the erosion of stratified layers. One user of the GOTHIC containment code using very coarse meshing obtained as good as the best CFD code predictions. However, the far worse results obtained by the other GOTHIC user imply that the success of such coarse-mesh approaches depends strongly on user experience.

The first general observation to be drawn is that even for such a simple, basic flow situation as the one set up in this exercise, with no complex physical processes taking place, a large spread of numerical predictions has been obtained. The key parameter - the rate of erosion of the stratified, helium-rich layer - was strongly overpredicted in some simulations, while other simulations predicted the persistence of the stratification to the end of the simulation time, itself an overestimate of the time for complete mixing observed in the experiment. This large spread in results, which became more obvious with elapsed time into the transient, indicates that there is still a need for additional learning to take place in the use of CFD in applications of this type.

The activity noted that CFD is no substitute for properly understanding the basic thermal-hydraulic phenomena involved in a numerical analysis being described. The CFD tools are useful to quantify the complex interplay between the various physical processes taking place rather than being given the burden of identifying them. The benchmark reinforced the need for an estimate of measurement uncertainty represented as error bars on experimental data because test measurements are almost worthless for validation purposes without this pre-requisite. The benchmark sent a strong request to have best practice guidelines on the presentation of experimental data.

The increased use of CFD codes to nuclear reactor safety application required a Best Practice Guidelines (BPG) and WGAMA issued a complete set of guidelines for a range of single phase applications of CFD to nuclear reactor problems (NEA, 2014a). The document was intended to provide direct guidance on the key considerations in known single phase applications, and general directions for resolving remaining details. The document listed a set of problems and their solution approach by isolating the portions of the nuclear reactor safety problem most in need of CFD and the use of a classic thermal-hydraulic safety code to provide boundary conditions for CFD. The document provided recommendations from experts on models associated with buoyancy, heat transfer, free surfaces, and fluid structure interactions. Guidance is given on convergence of iterative solutions, and numerical techniques for following free surfaces. One chapter was dedicated to approaches used in limiting errors associated with discretization and numerical verification methods.

Over the past ten years, the WGAMA initiated activities to promote the use of CFD for Nuclear Reactor Safety, which are delineated in the preceding paragraph. A list of safety issues for which CFD may bring real benefits was also established through several workshops and benchmarking exercises. All of these activities provided more confidence in the application of CFD for nuclear reactor safety by defining the conditions and requirements for establishing some confidence in the predictions. However, no applicable methods have been published about a possible quantitative evaluation of the uncertainty of predictions, and

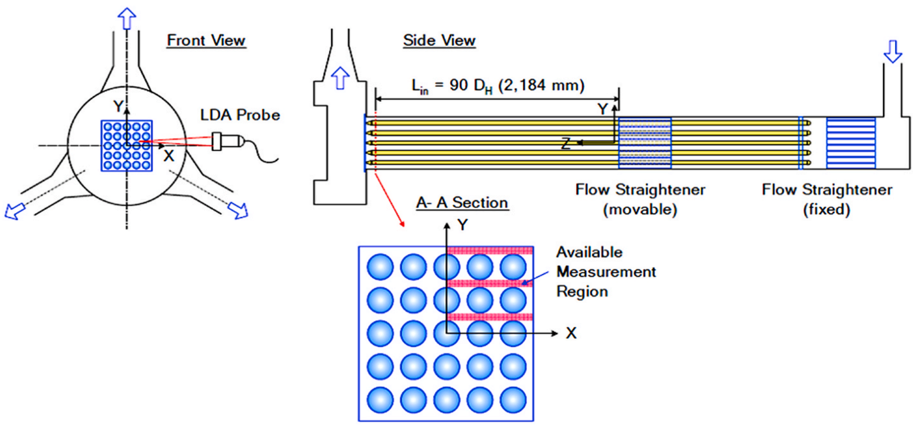


Fig. 8. Front (left) and side (right) view of MATiS-H test rig (a schematic of LDA region included).

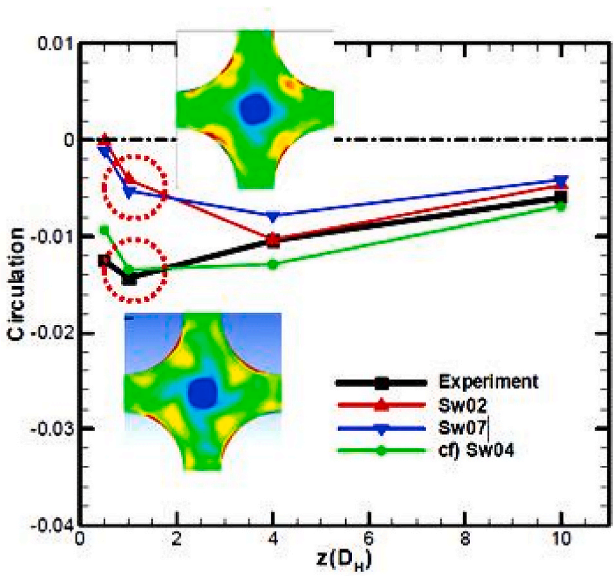


Fig. 9. Variation of circulation (swirl-type spacer design).

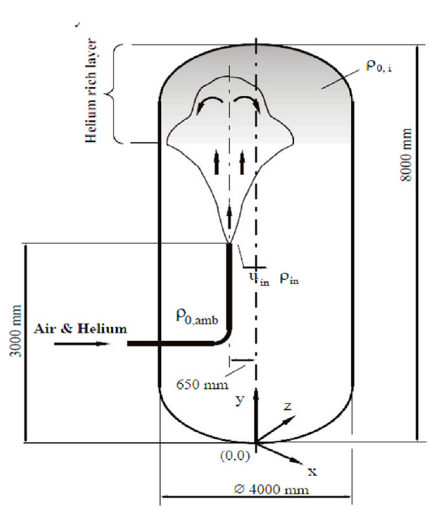


Fig. 10. The PANDA Test configuration used for the benchmark study of erosion of a helium-rich, stratified layer.

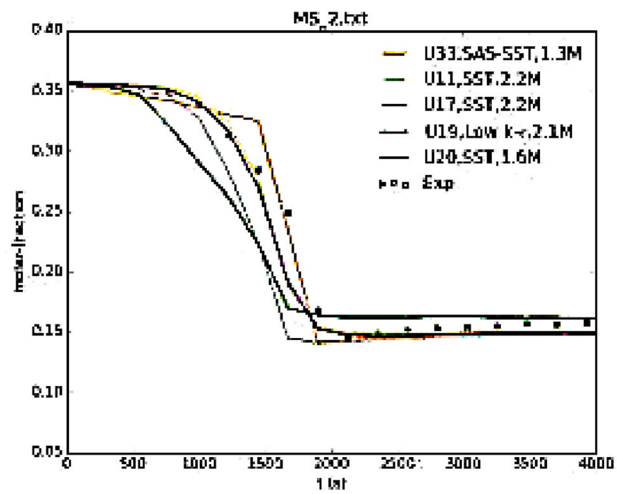


Fig. 11. Time-histories of helium concentration (molar fraction) at one of the uppermost location y = 6706 mm.

such an evaluation is mandatory for complementing a best estimate approach within a nuclear reactor licensing framework. Thus, a review of the methodologies for determining the uncertainty of CFD predictions applied to reactor thermal-hydraulics was initiated (NEA, 2016e). This is a very recent area of investigation, and the reported activity is rather limited. Only a few prospective activities are in progress. For this activity, one must compile a list of what exists in order to determine what more is needed. A comparison with system codes is useful since available uncertainty methods for system codes are rather mature as demonstrated in the BEMUSE project (NEA, 2011a).

A review of existing work in this field was conducted, but only very limited information was found on CFD Uncertainty Quantification (UQ) applied to nuclear reactor safety analysis. The main reactor issues for which CFD uncertainty quantification methods are expected to be applicable in the short and medium term are mixing problems (e.g. temperature, boron concentration, hydrogen concentration) with or without density effects. The two types of methods developed and used for UQ of system codes, such as the propagation of input parameter uncertainty and the extrapolation of output accuracy, are useful for CFD uncertainty quantification. The first method determines the uncertainty of all input uncertain parameters and propagates these uncertainties in the reactor calculation. The second method measures the accuracy of code predictions from separate effects tests simulating a reactor transient and extrapolates the accuracy to the reactor application.

Maturity of all uncertainty quantification methods is low and all of them need extensive testing and benchmarking. Although CFD UQ is still in its early stages, application of some existing methods – if properly done and well tested – seems achievable. The application of single-phase CFD to safety demonstration does not give rise to insurmountable difficulties, and such new technology may reach a degree of maturity comparable to that of system codes, at least for a few first applications in the short or medium term. The application of Best Practice Guidelines (BPG), a comprehensive assessment relative to the application, and a consolidated UQ method are the main requirements. A higher priority should be put on progress toward the latter criterion listed.

3.3. Severe accidents

As mentioned at the beginning of the section, the Fukushima-Daiichi accident shaped, to a large extent, the activities launched in this period, particularly those related to Status Reports (i.e., documents which compile the available information worldwide on specific topics). Three reports were produced closely related to severe accident management (SAM): on Filter Containment Venting Systems (FCVS), H_2 risk management and related code modeling, and loss of coolant accidents in Spent Fuel Pools (SFP); all these activities received a high level of participation.

The main driver of the report on FCVS was the decision made by many countries to implement FCVS in the Fukushima-Daiichi accident aftermath, where these were not yet applied at the time (NEA, 2014b). A first report on the issue had been already published in 1998. However, the evolution in knowledge and models to assess radioactive releases, the improvement of the filtration technologies and the possibility of enhancing existing venting strategies, indicated the need to compile updated information available on FCVS. The report provides a comprehensive description of safety requirements associated with FCVS and the status of implementation worldwide, together with the venting strategies for emergency operating procedures (EOPs) and SAM domains. It includes a description of the existing filtration technologies (i.e., scrubbers, deep-bed filtration and sorption systems) and the general FCVS design requirements. The effect of FCVS operation in NPP source term calculations is illustrated and, despite the expected benefits, potential negative aspects are also discussed. Possibilities for enhancement of already implemented systems are also discussed in the report.

The Fukushima-Daiichi accident confirmed once again the significant risk potential associated to H_2 during severe accidents. A report compiling all the relevant information on H_2 that might be useful in reviewing SAM strategies and even help in every decision related to H_2 mitigation, like installing H_2 measurement systems, was issued (NEA, 2014c). Right after the Fukushima-Daiichi accident, H_2 mitigation systems were required in most countries inside the containment (wherever such a requirement did not exist before) and some considered extending such safety safeguards outside the primary containment. In-containment, an overview of the systems used revealed that strategies are primarily containment-design dependent. The report also reviews the H_2 modeling strategies used worldwide. Most countries are using lumped parameter (LP) codes (e.g., integral or system codes with mechanistic models) for full plant long term SA analysis combined with 3D-like or 3D codes for detailed short-term and/or local hydrogen analysis (e.g., hydrogen distribution, combustion and mitigation). Integral or system codes are capable of calculating hydrogen generation in the reactor core and/or from MCCI in the cavity. Most codes have capabilities to model hydrogen distribution, combustion, mitigation systems and engineered safety features. In addition, some countries have developed more complex models with CFD codes for better assessing hydrogen combustion, recombination and key phenomena such as condensation or evaporation, which can affect hydrogen distribution in the containment. None of the codes are fully validated due to a lack of experimental data to cover the desired application range. Engineering judgement and a large degree of experience on code application is therefore needed in order to obtain realistic results. R&D will continue to reduce uncertainties and provide insights to refine the SAMGs.

No fuel degradation has been observed in SFPs during the Fukushima-Daiichi accident other than the one caused by external mechanical loads. Nevertheless, the peak temperatures in Unit 4 SFP reached not far from saturation (by just about 8 K) and only 1.5 m of water remained over fuel assemblies. Thus, although SFPs were demonstrated to be robust monolithic structures with redundant cooling systems connected to emergency back-up power, a status report was issued (NEA, 2015a) compiling comprehensive description of every aspect related to SFP safety. The report is focused on at-reactor SFPs and it identifies the most challenging events that would potentially threaten the system (loss of cooling and loss of inventory accidents). An overview of phenomena that would be expected in case of a severe accident, including the pure thermal-hydraulic phase and potential criticality issues, has been included. Even though there are analytical tools to model these scenarios, their application was considered to be not very straightforward, given that such tools were derived for reactor scenarios. The database review also showed that, even if some tests conducted could be applicable to SFPs, there existed at the time a few tests addressing specifically safety aspects of SFPs (OECD-SFP project; NEA, 2013a). The report contains several recommendations streamlining future WGAMA activities, like: a Phenomena Identification Ranking Table (PIRT) to focus further studies, a best practice guidelines (BPG) for a proper use of severe accident tools in the SFP domain, and a State-of-the-Art Report (SOAR) on SFP loss-of-cooling and loss-of-coolant accidents accounting for all the research conducted since the Fukushima-Daiichi accident.

In addition to the above Status Reports, other key activities were conducted in this period. An international Iodine workshop was organized in March 2015 in Marseille jointly by OECD/NEA, the NUGENIA association, the European Commission and IRSN. Generally speaking the workshop intended to assess the recent progress made on Source Term research and their application in AM. The essence of the conclusions and recommendations of the workshop regarding source term research and its implementation in tools supporting accident analysis and management including emergency response are detailed in a separate specific paper (Jacquemain et al., 2016). They mostly concern the necessity: (1) to perform additional research focused on reactor applications to improve the assessment of potential effects of "delayed" FP re-emission in SA from deposits in RCS, containment, solid filter surfaces and from pools (sumps, suppression pools, liquid pools in filters) on source term evaluations, (2) to deepen the assessment of the validity of source term related models implemented in SA system codes, and (3) to assess the various methods for source term evaluations and quantification of associated uncertainties. Full proceedings and a summary report of the workshop are available (NEA, 2016b).

A benchmark exercise was undertaken by WGAMA to investigate the current state of the art on fast running methods and tools for predicting the accident source term of radioactive releases and resulting public doses and to promote international cooperation in future development of such tools (NEA, 2015b). The project has demonstrated that the know-how for performing such fast, inevitably approximate accident modeling is quite advanced, benefitting from the mature understanding of the accident phenomenology, software and hardware advances as well as previous development effort in several organizations. Nevertheless, it has been shown that setting up even a relatively simple model to perform accident progression assessment may be a complicated task, especially if dealing with not-so-familiar reactor technology. The spread in predictions was substantial (Fig. 12), as explained by the varying capabilities of the tools, as well as by the assumptions made by the project participant regarding the possible accident progression. Based on the project results (NEA, 2015b), several recommendations were made: to follow-on work to achieve a more comprehensive understanding of differences in source term and dose results using fast-running software for emergency response; further study of source term, atmospheric transport, and dose calculation models in these software tools would benefit decision-making for formulating protective action recommendations; also, strategies for quickly predicting accident progression and projecting consequences could be a useful undertaking; finally, a forum for exchange of best practices and training for users of the software should be considered.

A State-Of-the-Art Report (SOAR) on Molten Corium Concrete Interaction (MCCI) and coolability was released in 2016 (NEA, 2016d; Basu et al., 2016). In the report, a concerted vision of the phenomenology of core-concrete interactions and melt coolability is summarized together with a global overview of simulation code capabilities and validation status. This concerted vision demonstrates the significant progresses made on the level of understanding regarding MCCI behaviour under both wet and dry cavity conditions but also led to identify a few issues (also based on lessons learned from Fukushima-Daiichi situations) that may warrant further investigation to reduce residual uncertainties. These issues include addressing realistic reactor configurations from the short to the long term, particularly with metal-rich melts, and improvement of top flooding melt coolability modeling. The report has been a pillar for the elaboration of the ROSAU research project which will address from 2019 these remaining issues.

Finally, as a follow-up activity of the previous TMI-2 benchmark exercise, an exercise was conducted to examine the code capability to predict core melt progression and the effect of SAM actions (NEA, 2015c). After a first phase in which comparisons to severe fuel damage experiments were set to confirm codes consistency, the work focused on modeling accident scenarios with SAM actions in place (particular attention was paid to core reflooding), so that the codes might get credit to even optimize the use of available cooling water sources. It was observed that before significant core degradation takes place, the code responses were mostly consistent. The phenomena controlling the event timing in the early phase degradation were identified to be: clad rupture, clad oxidation runaway and clad failure, and relocation of molten Zr. During and after substantial core melting and relocation, the divergence in code results became more pronounced (it was noted that user guidelines and additional experiments would help to reduce the spread observed). In short, differences in molten-material slumping and corium behavior in the lower head modeling highlighted the need for further exploration. One of the phenomena that carries most of the uncertainty is the reactor pressure vessel failure which time was shown to be very sensitive to failure criteria applied and models of corium behavior in the lower plenum, which was stated to need further validation and benchmarking.

4. Recent finalized activities (2017–2019)

4.1. Thermal-hydraulics

The NEA has been conducting a series of thermal-hydraulic experimental programs through joint collaborations among interested member countries since 2001 on a number of facilities such as SETH,[3] PKL.[4] These projects produced data in the integral test facilities to benchmark system codes for current and new PWR design concepts. Subsequently in 2014, ATLAS facility also joined the group of integral test facilities. In particular, ATLAS has been used to focus on design extension condition (DEC) scenarios to validate safety analysis codes. Since all of these facilities have a common goal, a joint workshop was organized and it attracted 60 participants from 16 countries (NEA, 2017a). The activity provided an efficient forum to evaluate the current code capabilities for the scenarios conducted in the projects.

As part of the workshop activity, three different counterpart tests were conducted within the current PKL3 and ATLAS projects. One of the recommendations provided by the Core Exit temperature Task Group was to have follow-up activities within various facilities to perform counterpart test to address the Core Exit Temperature (CET) issue (NEA, 2010). The calculations in this activity also confirmed earlier experience that system codes tend to underestimate the delay time between PCT and the CET, where AM actions are usually taken when a certain superheating is detected at the CET in PWRs worldwide.

The experience gained from conducting counterpart tests was recognized as a valuable exercise in understanding the underlying phenomena and in enabling improvements to scaling techniques. These counterpart tests produced consistent experimental results between LSTF and PKL, and LSTF and ATLAS facilities. The so called “cliff-edge effect” during the IBLOCA scenario was highlighted by comparing experimental database from LSTF and ATLAS. There were, however, some identified differences in the results among facilities, and the application of scaling methods has been recommended to clarify the root cause of these differences. The code results have shown good agreement with experimental results and it is indicative of the ability to simulate complex phenomena in the current thermal-hydraulic codes. The use of blind calculations within the benchmarks confirmed to be a useful method to test predictive capabilities of thermal-hydraulic codes. Some of the code applications have also used sensitivity and uncertainty analyses as a part of these blind exercises.

Similar to past experiences, bringing specialists from the analytical and experimental fields together provided significant developmental opportunities to both groups. The interaction has helped identify code and experimental deficiencies. Sharing of modelling justifications and the reasons for deviations from experimental data allows healthy debate on what improvements are required, either in the models or in the experimental techniques, enhancing cross-fertilization among experts. For example, the relative importance of ambient heat losses in transients at nuclear power plants (NPP) in relation to experimental facilities was confirmed by the experts: as it is well known, the surface area to volume ratio in NPPs is relatively small compared to experimental facilities and therefore the ambient heat losses are often ignored in NPP simulations, whereas it has an effect on the experimental facilities. For long transients, the impact of ambient heat losses on NPPs may turn out to be significant and definitely warrant some consideration in thermal-hydraulic analysis.

Activities started in 2016 dealing with passive systems or better the thermal-hydraulics of passive systems with focus on natural circulation. The concerned working group is expected to issue a report in 2020. As a key relevant finding at the moment is the observation that the passive systems are not the panacea for nuclear reactor safety. Design and introduction of passive systems into the design of nuclear reactors may or may not improve the overall safety: specific activities dealing with ‘reliability of phenomena evolutions’ must be considered together with the uncertainties which may be embedded into the predictions of transient passive system performances by current numerical tools.

4.2. CFD applications in nuclear safety analyses

The promotion of CFD for nuclear reactor safety applications continued in the 2017 to 2018 period with a WGAMA workshop, CFD4NRS-6 (NEA, 2017b) and CFD4NRS-7. The CFD4NRS-7 workshop took place in 2018 and the summary document has not yet been produced. As a result, this workshop will not be discussed here. In the CFD4NRS-6 workshop, the organizers made a particular observation that there was significant evidence and progress in the adoption of CFD for nuclear safety related applications. These nuclear safety related applications, discussed in the workshop, ranged from nuclear fuel to containment.

The WGAMA benchmarks have been noted to engage a larger sector of the international community. With this extended engagement of participants, nuclear fuel sub-channel flow related applications have reached a good level of maturity, as demonstrated by successful international blind and open benchmarks, mainly due to CFD-grade

[3] SESAR Thermal Hydraulics project and SESAR represents Senior Group of Experts on Nuclear Safety Research.

[4] Primary Coolant Loop Test Facility Project.

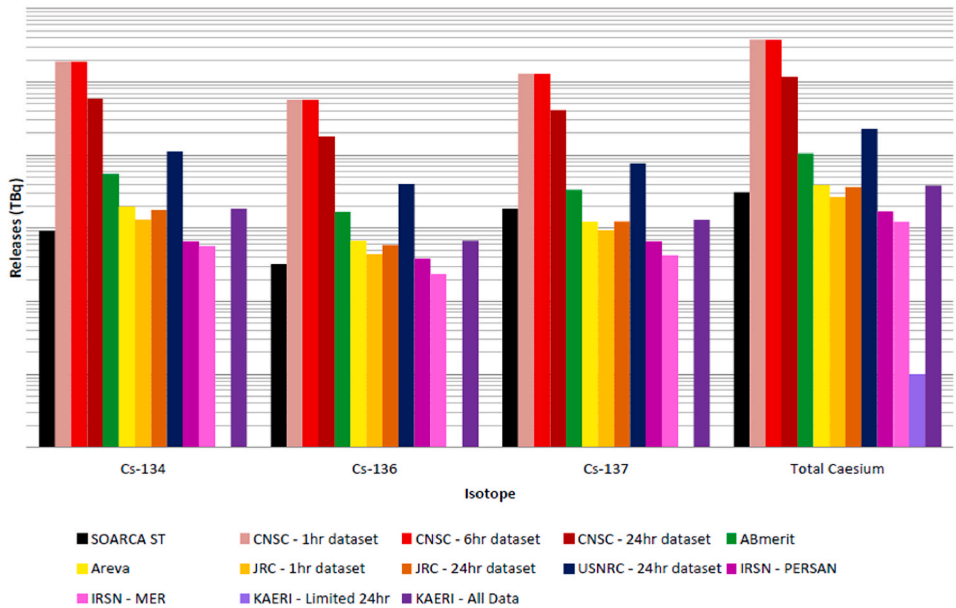


Fig. 12. Estimated Cs release for an unmitigated long-term Station Black-Out in a PWR.

experiments. As a follow-up to previous containment benchmark on erosion of a helium-rich, stratified layer (NEA, 2016c), a large number of high quality papers is indicative of the passion researchers have towards improving and understanding the phenomena in order to provide valuable guidelines for CFD application. As CFD methods mature, advances in uncertainty quantification to support reactor licensing is required. The GEneric Mixing eXperiment (GEMIX) blind benchmark exercise, discussed below, represented a good first step and focused on some of the fundamental challenges the CFD community will be addressing in the near future. The workshop also combined the blind CFD benchmark activity with keynote lectures, a plenary lecture, a poster session on GEMIX benchmark, and fourteen sessions of contributed paper presentations. The model was very much appreciated by the participants who provided the possibility to display their work as posters rather than papers and discuss their experiences.

The main objectives of a benchmark exercise on a mixing layer test in the GEMIX facility was to promote the evaluation of various uncertainty quantification methodologies for computational fluid dynamic (UQCFD) applications for nuclear reactor safety (NRS). The assessment of the uncertainty quantification (UQ) methodologies was based on a benchmark code assessment performed against a blind experiment (NEA, 2017c). The numerical benchmark exercise was restricted to single-phase flow, with turbulent mixing in the presence of density gradients, which is a typical situation encountered in many reactor issues, where CFD is currently used. The exercise was based on an experiment performed in the GEMIX facility in Paul Scherrer Institute, Switzerland. A simplified schematic of the flow mixing arrangement is shown on the left hand side in Fig. 13. In the experiment, two turbulent co-flowing water streams of equal velocity but different density were allowed to mix downstream of a splitter plate inside a square mixing section. The density of the lower stream is increased by adding sucrose in conjunction with a temperature adjustment. The velocity and concentration field measurements were obtained when a stably stratified condition established as the denser stream located below the lighter stream. The experiment was conducted at unstratified conditions to obtain the reference condition in order to quantify the influence of different density stratifications. The participants submitted a calculation for the blind test case, where they presented their predictions for mean velocity, turbulence kinetic energy and concentration profiles. All the results included uncertainty bands. Since some methodologies for UQCFD use data from a validation step (for the definition of the model, its calibration and/or extrapolation of errors), three open tests cases were provided to the participants. It is noteworthy to mention that for the blind test, the density ratio between the two mixing streams was 1%, which is much lower than the values encountered, for example, in pressurized thermal shocks. Among twenty calculations three calculations predicted the mixing layer thickness exceptionally well as shown in Fig. 13 (right hand side).

The uncertainty propagation method and combined accuracy extrapolation and uncertainty propagation methods provided better results in the blind simulations. The participants using a combined method (propagation and extrapolation) obtained the best agreement with the blind data. The benchmarking activity indicated that the most important step in the UQCFD analysis is proper characterization of the input uncertainties.

4.3. Severe accidents

In the severe accident field, WGAMA activities have been targeted at enhancement of management of severe accidents, including on the long term, integrating knowledge gained through recently concluded research projects, through development of analytical tools and through on-going management and analyses of the Fukushima-Daiichi accident. Related to the Fukushima-Daiichi accident, tight connections and knowledge sharing have been recently organized between WGAMA and CSNI post-Fukushima-Daiichi projects addressing analyses of the accident (BSAF) and preparatory actions for the damaged fuel retrieval and decommissioning operations (ARC-F, PreADES).

A report addressing how to inform SAMG and actions through analytical simulations has been released in 2017 (NEA, 2017d). It describes the diverse current practices related to SAMG verification and validation including expert judgement, simulators, field training, tabletop exercises, emergency drills and exercises and analyses. It addresses more specifically the use of analytical simulations including the impacts of operator actions on accident progression to inform the SAMG developers, users and regulators. The general guidance and information provided in the report should be useful to utility personnel involved in verification and validation of SAMG, as well as to regulatory staff performing evaluation of generic or plant-specific SAMG.

Following the status report on SFPs under loss of coolant accident conditions (NEA, 2015a), a PIRT has been conducted (NEA, 2017e) to recommend needed research actions to enhance analytical simulation

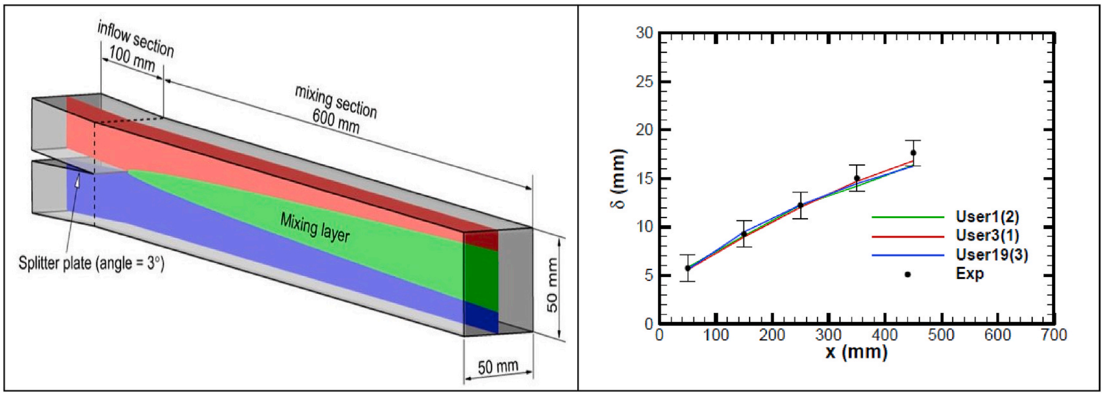


Fig. 13. The GEMIX experimental schematic (left) and results from three select blind simulations (right).

tools and mitigation approaches for both loss of cooling and loss of coolant accidents in a SFP. Experimental investigations have been suggested to better address (1) cladding oxidation in mixed air-steam atmospheres, (2) potential for cooling partially uncovered fuel assemblies, (3) natural recirculation patterns at the pool scale and (4) potential for cooling by spray. The report also recommends developing sensitivity and uncertainty analyses for such accidents and to elaborate a SOAR in the near future.

To complement the MCCI SOAR (NEA, 2016d), a status report on ex-vessel steam explosion has been produced (NEA, 2017f) to identify remaining issues related to assessing the steam explosion risk in connection to ex-vessel corium cooling strategies. The report recommends performing further analytical experimental investigations regarding oxidation and solidification of single melt drops, the fragmentation of prototypical drops in reactor conditions and film boiling at very high temperature as well as more integral experiments to investigate realistic reactor configurations (e.g., pressurized and inclined melt jets, partial melt stratification, lateral jets impacting on walls).

A status report on long term management and actions for a severe accident in a NPP has been recently released (NEA, 2018). The report provides recommendations to enhance long term management of a severe accident and for future related research largely based on a compiled feedback from long term management of major past-accidents (TMI2, Chernobyl and Fukushima-Daiichi). Needed provisions enhancement relate in particular to the optimization of management of cooling waters, to the limitation of radioactive releases due to remobilization phenomena on the long term and to maintaining cooling and sub-criticality of the damaged fuel on the long term. Further research should address cross-cutting issues such as knowledge development or consolidation for (1) status of components, equipment, systems and structures critical for maintaining safety functions including the long term, (2) long term phenomena (corrosion-erosion reactions, clogging of recirculation loops, degraded fuel leaching and ageing, fuel dusting and dispersion), (3) methods or systems for risk assessment for long term management actions optimization (including e.g., re-criticality, fuel dusting and dispersion in fuel recovery operations).

An international conference on SAMG enhancement after the Fukushima-Daiichi accident has been organized under the WGAMA auspices by CNSC in Ottawa in October 2018. The conference has been a large success with participation of about 200 delegates from 21 countries. It covered: (1) post-Fukushima Daiichi enhancements of the SAM requirements, principles, strategies and procedures; (2) equipment for SAM; (3) human factors under accident conditions and (4) use of R&D results in strengthening accident management effectiveness. Conclusions and recommendations from the conference will be published by the end of 2019.

Finally, an international workshop to address remaining issues in the source term area was held in January 2019 in Paris in NEA premises. Conclusions and recommendations from the workshop are expected at the end of 2019 which should guide the elaboration of proposals for future research projects in the area after the completion of the on-going BIP3, THAI3 and STEM 2 projects in 2019. It has been recognized, in particular through the analyses of the Fukushima-Daiichi accident, that phenomena resulting in remobilization of radionuclides contributing to the radiological consequences on the short and longer terms deserve further attention.

5. Final remarks

A summary of the industrious production of WGAMA in the area of analysis and management of accident in the last 20 years has been outlined. The text is loaded with references in which substantial support may be found. As a result, around 50 technical reports (status reports and state of the art reports, workshop proceedings, code benchmarking reports, uncertainty quantification reports, best practice guidelines, and expert opinion papers) have been cited at the end of this paper.

The research activities have evolved based on emerging issues in the reactor safety area and consensual interest of the working group members. Despite the areas covered by WGAMA were defined about 20 years ago, there has been a continuous update according to the NEA member countries' concerns. It is worth highlighting the outstanding capability that the group has demonstrated to address tasks related to unexpected events, like the Fukushima-Daiichi accident, and how the Group has been able to shape up their activities to cope with that circumstance without neglecting the working group's mandate, strategic initiatives and scheduled activities.

WGAMA keeps as active as in the two past decades and a number of activities have just been or are about being launched. Just to mention a few recently initiated activities encompassing the three domains in WGAMA scope: a critical assessment of passive safety systems designs and safety assessment and recommendations for further research in the area, if needed, are expected by 2020; a new edition of the workshop on experimental validation and application of CFD and Computational Multi-Fluid Dynamics (CMFD) codes to Nuclear Reactor Safety Issues (CFD4NRS-8) is being planned and will be held in Paris-Saclay during September 2020; a SOAR compiling the knowledge gained through research and safety studies concerning the management of H_2 and CO combustion risk in a NPP is expected to be published by the end of 2021, including the suggestions coming out in terms of gaps being worth of further investigation.

Phenomena and phenomenological knowledge based on experiments constituted the foundation of CSNI activities in the areas of WGAMA since the establishment of cooperative research programs in the late 70's. This will continue in the future: measured phenomena are the bricks and the glue to construct the knowledge, namely, to validate complex procedures and numerical tools which allow the safety demonstration of nuclear reactors. This will continue to streamline the mandate and future activities of the group.

Definitely, the WGAMA continues recommending further research on those safety issues closely involved in the analysis and management of accidents that have the potential to substantially enhance the current capabilities existing worldwide to better cope with such events.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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References

Basu, S., Bonnet, J.-M., Cranga, M., Vola, D., Farmer, M.T., Robledo, F., Spengler, C., 2016. Learnings from 30 Years of R&D on molten core concrete interaction. In: The 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety, (NUTHOS-11), Gyeongju, Korea, October 9-13, 2016.
Jacquemain, D., Albiol, T., Dickinson, S., Herranz, L.-E., Funke, F., Glowa, G., Gupta, S., Hoshi, H., Hotta, A., Kärkelä, T., Kissane, M., Lind, T., Salay, M., Song, J.-H., Van Dorsselaere, J.-P., 2016. Conclusion of the international OECD/NEA-NUGENIA iodine workshop. In: The 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety, (NUTHOS-11), Gyeongju, Korea, October 9-13.
NEA, 2000. ISP-43: Rapid Boron Dilution Transient Experiment, Comparison Report, vol. 22. NEA/CSNI/R(2000), 2000.
NEA, 2001a. AECL International Standard Problem ISP-41 FU/1 Follow-Up Exercise (Phase I): Containment Iodine Computer Code Exercise Parametric Studies, vol. 17. NEA/CSNI/R(2001), 2001.
NEA, 2001b. Validation Matrix for the Assessment of Thermal-Hydraulic Codes for VVER LOCA and Transients, A Report by the OECD Support Group on the VVER Thermal-Hydraulic Code Validation Matrix, vol. 4. NEA/CSNI/R(2001), 2001.
NEA, 2002a. Proceedings of an Exploratory Meeting of Experts to Define an Action Plan on Best Estimate Calculations and Uncertainty Analysis, vol. 15. NEA/CSNI/R (2002). Aix en Provence, 13-14 May 2002, 2002.
NEA, 2002b. Proceedings of an Exploratory Meeting of Experts to Define an Action Plan on the Application of Computational Fluid Dynamics (CFD) to Nuclear Reactor Safety Problems, vol. 16. NEA/CSNI/R(2002). Aix-en-Provence, 15-16 May 2002, 2002.
NEA, 2002c. ISP-45 (QUENCH 06 Test) - Final Comparison and Interpretation Report, vol. 23. NEA/CSNI/R(2002), 2002.
NEA, 2003. ISP-44 - Comparison and Interpretation Report - Final, vol. 5. NEA/CSNI/R (2003), 2003.
NEA, 2004a. AECL International Standard Problem ISP-41 FU/2 Follow-Up Exercise (Phase II): Iodine Code Comparison Exercise against CAIMAN and RTF Experiments, vol. 16. NEA/CSNI/R(2004), 2004.
NEA, 2004b. ISP-46 - PHEBUS FPT-1 Integral Experiment on Reactor Severe Accidents, vol. 18. NEA/CSNI/R(2004), 2004.
NEA, 2006. BEMUSE Phase 2 Report: Re-analysis of the ISP-13 Exercise, Post Test Analysis of the LOFT L2-5 Test Calculation, vol. 2. NEA/CSNI/R(2006), 2006.
NEA, 2007a. SOAR on Iodine Chemistry, vol. 1. NEA/CSNI/R(2007), 2007.
NEA, 2007b. Proceedings of the Workshop on Benchmarking of CFD Codes for Application to Nuclear Reactor Safety (CFD4NRS), vol. 3. NEA/CSNI/R(2007). Munich, 5-7 September 2006, 2007.
NEA, 2007c. Best Practice Guidelines for the Use of CFD in Nuclear Reactor Safety Applications, vol. 5. NEA/CSNI/R(2007), 2007.
NEA, 2007d. International Standard Problem ISP-47 on Containment Thermal Hydraulics: Final Report, vol. 10. NEA/CSNI/R(2007), 2007.
NEA, 2008. BEMUSE PHASE IV Report: Simulation of a LB-LOCA in ZION Nuclear Power Plant - Main Report, vol. 1. NEA/CSNI/R(2008), 6/, 2008.
NEA, 2009a. Ability of Current Advanced Codes to Predict Core Degradation, Melt Progression and Reflooding - Benchmark Exercise on an Alternative TMI-2 Accident Scenario, vol. 3. NEA/CSNI/R(2009), 2009.
NEA, 2009b. SOAR on Nuclear Aerosols, vol. 5. NEA/CSNI/R(2009), 2009.
NEA, 2009c. "Proceedings of the Workshop on Experiments and CFD Codes Application to Nuclear Reactor Safety (XCFD4NRS)", Grenoble, France, 10-12 September 2008, vol. 12. NEA/CSNI/R(2009), 2009.
NEA, 2009d. BEMUSE Phase V Report: Uncertainty and Sensitivity Analysis of a LB-LOCA in ZION Nuclear Power Plant, vol. 13. NEA/CSNI/R(2009), 2009.
NEA, 2010. Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor, vol. 9. NEA/CSNI/R(2010), 2010.
NEA, 2011a. BEMUSE Phase 6 Report - Status Report on the Area, Classification of the Methods, Conclusions and Recommendations, vol. 4. NEA/CSNI/R(2011), 2011.
NEA, 2011b. Proceedings of the Workshop on CFD for Nuclear Reactor Safety Applications (CFD4NRS-3), vol. 14. NEA/CSNI/R(2011). Bethesda, USA, September 14-16 2010, 2011.
NEA, 2011c. ISP-49 on H_2 Combustion, vol. 9. NEA/CSNI/R(2011), 2011.
NEA, 2012. International Standard Problem No.50 on ATLAS, ume 1 to 3. NEA/CSNI/R (2012), 6, 2012.
NEA, 2013a. Seminar of the SFP Project – Executive Summary, vol. 13. NEA/CSNI/R (2013).
NEA, 2013b. Report of the OECD/NEA-KAERI Rod Bundle CFD Benchmark Exercise, vol. 5. NEA/CSNI/R(2013), 2013.
NEA, 2014a. "Best Practice Guidelines for the Use of CFD in Nuclear Reactor Safety Applications – Revision, vol. 11. NEA/CSNI/R(2014), 2014.
NEA, 2014b. Status Report on Filtered Containment Venting, vol. 7. NEA/CSNI/R(2014), 2014.
NEA, 2014c. Status Report on Hydrogen Management and Related Computational Codes, vol. 8. NEA/CSNI/R(2014), 2014.
NEA, 2015a. Status Report on Spent Fuel Pools under Loss-Of-Coolant Accident Conditions Final Report, vol. 2. NEA/CSNI/R(2015), 2015.
NEA, 2015b. Benchmark of Fast-Running Software Tools Used to Model Releases during Nuclear Accidents - Final Summary Report, vol. 19. NEA/CSNI/R(2015), 2015.
NEA, 2015c. Ability of Current Advanced Codes to Predict In-Vessel Core Melt Progression and Degraded Core Coolability - Benchmark Exercise on Three Mile Island-2 Plant - Final Report, vol. 3. NEA/CSNI/R(2015), 2015.
NEA, 2016a. Scaling in System Thermal-Hydraulics Applications to Nuclear Reactor Safety and Design: a State-Of-The-Art Report, vol. 14. NEA/CSNI/R(2016), 2016.
NEA, 2016b. "International Iodine Workshop", Marseille, France, March 31 – April 1, 2015, Full Proceedings, vol. 5. NEA/CSNI/R(2016), 2016.
NEA, 2016c. State-of-the-Art Report on Molten-Corium-Concrete Interaction and Ex-Vessel Molten-Core Coolability, vol. 15. NEA/CSNI/R(2016), 2016.
NEA, 2016d. The Nuclear Energy Agency/Paul Scherrer Institute Computational Fluid Dynamics Benchmark Exercise, vol. 2. NEA/CSNI/R(2016), 2016.
NEA, 2016e. Review of Uncertainty Methods for Computational Fluid Dynamics Application to Nuclear Reactor Thermal Hydraulics, vol. 4. NEA/CSNI/R(2016), 2016.
NEA, 2017a. Summary and Conclusions of the Joint PKL-3-ATLAS Workshop on Analytical Activities Lucca, Italy 13-15 April 2016, vol. 10. NEA/CSNI/R(2017), 2017.
NEA, 2017b. Computational Fluid Dynamics for Nuclear Reactor Safety Applications-6 (CFD4NRS-6) Workshop Proceedings 13-15 September 2016 Cambridge USA", vol. 20. NEA/CSNI/R(2017), 2017.
NEA, 2017c. OECD/NEA CFD–UQ Benchmark Exercise: CFD Prediction and Uncertainty Quantification of a GEMIX Mixing Layer Test, vol. 19. NEA/CSNI/R(2017), 2017.
NEA, 2017d. Informing SAMG and Actions for NPP through Analytical Simulations, vol. 16. NEA/CSNI/R(2017), 2017.
NEA, 2017e. PIRT on SFP under Loss of Cooling and Loss of Coolant Accident, vol. 18. NEA/CSNI/R(2017), 2017.
NEA, 2017f. Status Report on Ex-Vessel Steam Explosion, vol. 15. NEA/CSNI/R(2017), 2017.
NEA, 2018. Status Report on Long Term Management and Actions for a SA in a NPP, vol. 13. NEA/CSNI/R(2018), 2018.
Smith, B.L., Mahaffy, J.H., Angele, K., Westin, J., 2011. Report of the OECD/NEA-Vattenfall T-Junction Benchmark Exercise, vol. 5. NEA/CSNI/R(2011), 2011.