**Advances in the analysis and management of nuclear power plant accidents and future challenges:**

**the OECD/NEA WGAMA and the Joint Nuclear Safety Research Projects**

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Abstract – *The Working Group on Analysis and Management of Accidents (WGAMA) of the OECD Nuclear Energy Agency (NEA) addresses activities in the three technical fields of thermal-hydraulics (T/Hs), computational fluid dynamics (CFD) and severe accidents (SAs) related to safety aspects of potential accidental situations in nuclear power plants (NPPs). The WGAMA assesses and strengthens the technical basis needed for the prevention, mitigation and management of potential accidents in nuclear reactors and related technologies, and facilitates international convergence on safety issues, safety assessments methods and accident management (AM) measures and strategies. This paper aims to review and summarize the recent WGAMA activities and outcomes by focusing on T/H analysis of water-cooled nuclear reactors and possible applications to advanced designs, featuring CFD applications to nuclear reactor safety (NRS).*

*This paper also describes benefits from, and opportunities for, the NEA Joint Safety Research Projects (JPs) which aim to enhance the technical bases for accident analyses with validation of T/H and CFD computer codes, with performing series of separate-effect tests (SETs) and/or integral-effect tests (IETs). The JPs contribute further to the development and preservation of key technical capabilities, research infrastructure and expertise in participating organizations/countries, and to education and training of the future generation of nuclear safety experts, both with WGAMA cooperation.*

**Keywords:** WGAMA, thermal hydraulics analysis, joint safety research projects

I. Introduction

The OECD/NEA is an intergovernmental agency that facilitates co-operation among countries with advanced nuclear technology infrastructures to seek excellence in nuclear safety, technology, science, environment and law. Its framework of standing technical committees, joint international undertakings, and Secretariat-serviced, separately-funded bodies allows the Agency to be flexible and responsive. Through its 60 years of international service, the NEA scientific and technical work has been at the forefront of knowledge. The NEA publishes consensus positions on key issues to provide credible references and examples of best practice based on the work of over 70 working parties and expert groups. The NEA JPs and information exchange programmes also allow interested members and non-members to join forces in carrying out international cooperating research programs on a cost-sharing basis.

**Working groups** (WGs) carry out the programme of work that includes various activities and that usually extends over a number of years. The **WGAMA** [1 - 3], among 11 WGs of the NEA Committee on the Safety of Nuclear Installations (CSNI), undertakes activities related to potential design-basis accidents (DBAs) and beyond design-basis accidents (BDBAs), including design-extension conditions (DECs) in NPPs and related technologies, especially on the safety aspects of operating reactors as well as emerging challenges of evolutionary and innovative reactor designs, including SMRs. In setting its own priorities, the WGAMA takes into account the needs of its members through their annual country reports, activity synthesis reports (e.g., SOARs, TOPs, benchmarks/ISPs, PIRTs, conference/ workshop summary reports, project summary application reports), relevant activities carried out by other CSNI WGs and Nuclear Science Committee (NSC) expert groups, the OECD/NEA JPs, and complementary activities at international organisations such as the IAEA and the European Commission. Over 100 members are participating in the WGAMA from 28 countries. Over 400 experts are involved in its activities, as both contributors and beneficiaries of the outcomes. Consequently, the achievements by the WGAMA are outstanding in the number and quality of technical reports and position papers as reference publications (CSNI or NEA reports), and in organizing workshops (WSs) and conferences that cover current and innovative technologies. In addition, the WGAMA aims to transfer the acquired knowledge to younger experts through the THICKET seminar series. This paper focuses on the review of the recent activities and outcomes from T/H analysis of water-cooled nuclear reactors and possible applications to advanced designs, especially on the CFD applications to NRS.

**NEA Joint safety research projects (JPs)** [4] are collaborative ventures governed by an agreement between a number of participating organisations. With close to 50 projects completed since 1958, they offer a flexible framework where different stakeholders (regulators, industry, TSOs, research organizations) join forces to address key safety issues and develop shared knowledge base for nuclear safety applications. Each JP is conducted by the organization(s) operating test facilities. In addition to providing key quality data sets to support safety analyses, the JPs contribute to the development and preservation of key technical capabilities, research infrastructure and expertise in participating organizations, and to education and training of the future generation of nuclear safety experts. This paper describes benefits from and opportunities for JPs in the T/H field.

II. Recent WGAMA activities in the field of T/H

The NEA CSNI organized the first Specialist Meeting (SM) on Transient two-phase Flows at Toronto in 1976. The SM on Nuclear Thermal-Hydraulics (NTH) has been successively organized by the NEA CSNI providing up-to-date T/H research result addressing continuous improvements of nuclear safety. Following such a historical effort, the WGAMA organized a specialists meeting on transient NTH in water cooled reactors (SM-TH) on March 2023. [5, 6] The event marked another historical step summarizing the whole status of current NTH that may advance basis of reactor safety assessment as above, along with the efforts through WGAMA international collaborative activities. Over 60 technical papers and 11 summary papers in 12 sessions summarized the status of NTH and identified key needs in safety research on T/H responses during reactor accidents.

The NTH activities nowadays highly rely on a multi-topics feature where multi-scale and multi-physics analyses and modelling efforts take a significant part. T/H efforts in experiments, modelling, code development, validation and application have served for such scientific and technological advancements during the last two decades. This includes best estimate plus uncertainty (BEPU) and CFD for reactor safety assessment, code coupling among different physics, connection with probabilistic safety assessment (PSA), artificial intelligence (AI) and machine learning (ML). All of these may form a sort of virtual layers between NTH and nuclear reactor applications. Any statement about current status and future development should consider this situation.

As such, the WGAMA has contributed to establish fundamental technical bases that critically serve for T/H safety analyses of operating LWRs. Perspectives include the assessment of the applicability of the established technical bases for SMRs, featuring scaling aspects and efficiency of passive systems.

III. Recent WGAMA activities and outcome in the field of CFD

Computational fluid dynamics (CFD) code is a cross cutting analysis tool potentially applicable to detailed analyses of local multi-dimensional flows (e.g., coolant circuits, fuel bundles, fuel melts, steam and/or gas that may convey fission product (FP) aerosols, and their mixtures) that may appear in all kinds of reactor accidents of DBAs, DECs, and SAs.

The WGAMA formed the CFD task group (TG) in 2000, which has actively produced several reference documents [7, 8]. The Best Practice Guidelines (BPGs) describes all the possible issues related to a nuclear safety study by means of CFD analyses [9, 10]. The CFD-TG has organized a series of CFD4NRS WS at two-year interval [11]. The ninth WS (CFD4NRS-9) was successfully held in February 2023, hosted by Texas A&M University in the United States [12]. The next one is planned to be hosted by JAEA in Japan tentatively in 2025. The CFD-TG has also organized six benchmark exercises since 2011 [13, 14]. The latest one was just started in 2023 on unstable thermal mixing and pipe wall fatigue in a dead-end leg connected to main piping with a T-junction.

A few examples of recent CFD-relevant activities are reported below.

***III.A. Writing tasks***

**Technical Opinion Paper (TOP) on the Use of CFD for Nuclear Safety** [7]: CFD resolves a higher level of phenomenological details compared to the established system T/H codes that are used for reactor safety assessment as system-scale tools. While the CFD appears promising, it is still not mature, especially in handling gas-liquid two-phase flows. CFD is thus limited to a relatively small number of applications, which raises questions about the evaluation and integration of CFD-based safety studies. The TOP aims to provide a clear picture of the current use and capabilities of the CFD with perspectives on main challenges hindering a larger use of CFD in NRS studies.

The challenges that have been identified to further extend the CFD use in NRS studies may arise from gaps in the established methodologies (e.g., Uncertainty Quantification (UQ)), the availability and access to an experimental database for code development and validation or the insufficient knowledge of CFD capabilities and/or limits outside of the experts’ community. Analyses of possible ways to overcome those challenges identified potential collaborative activities and led to recommendations of following priority activities for an extended use of CFD in NRS studies that will be pursued in the CFD-TG ongoing activities:

* Build a library of links for CFD-for-NRS-related data, including links to validation databases and to fundamental documents (e.g., BPG, State-of-the-Art Reports (SOARs)).
* Enhance reliability and credibility through blind CFD-model benchmarking, extended towards application-oriented comparative studies.
* Update and promote existing reference documents (like BPG, synthetic reports on CFD activities…).
* Support future work on the development of UQ methodologies for CFD. Collect and summarize existing works and organize exercise of application.
* Provide help to newcomers to CFD in a form of a few short, simple documents like “CFD for deciders”, “CFD for system code users”, “System codes for CFD users”.

**Best Practice Guidelines (BPGs)** **for the use of CFD in Nuclear Reactor Safety (NRS) Applications, revision** [9]: This is the second revision of the previous 2014 revision [10]. The report gives an overview of the current status of the CFD applicability and addresses future issues (e.g., parallelism and coupling with system codes). Some parts of the report should be useful for certain discussions on SMRs, such as those concerning containment wall condensation, natural convection, water hammer and liquid metal systems other than verification and validation (V&V) and phenomena identification and ranking table (PIRT) processes.

This 2nd revision of BPG serves as a basis for several issues regarding the use of CFD for NRS applications. It can be used, first of all, by specialists to build a rigorous methodology for a CFD-based study, and can also be used as a guide for the assessment of the CFD-based NRS study.

At the beginning of this BPG, a “Quick Guide Introduction” is provided as the most helpful to CFD beginners, by giving them the essential guidelines without having to search these in the entire BPG document.

As reflected in the document, producing a CFD analysis of a NRS issue implies much more than just generating input data and taking note of calculation results. The main recommendation is to clearly include documentation at each of all the steps in the corresponding NRS study. Trusting the CFD simulation results implies several considerations must be addressed. Moreover, even though the equations solved come close to describing the elementary physics of fluid flows (at least closer than in more global approaches), a set of models is still required, in particular for turbulence for single phase flows, which needs justifications and analysis of applicability. According to the relatively high computational cost of CFD numerical simulations in industrial configurations, some compromises may have to be considered regarding the accuracy of the solution by optimizing the mesh size; for example, this has to be compensated by a corresponding analysis of induced errors. For the specific point of UQs, the BPG provides some generic discussions regarding the issue and references. Current state of the art of CFD studies, however, reveals that methodologies mainly issued from system scale studies still require some adaptation when they are used without inducing excessive computational efforts in “real” industrial problems.

A future perspective for this activity is a regular updating of the document according to the most recent progress in the field of CFD as well as a potential transformation of the BPG into specific applications.

***III.B. Benchmark***

The current benchmark addresses **thermal mixing and fatigue in a dead-end leg connected to a main pipe with a T-junction** led by the Vattenfall for the experimental part (Vattenfall T-Junction facility) and by the PSI and ETZH for the analytical part.

The benchmark is aimed to assess and validate the CFD predictive capabilities in the evaluation of the failure risk of a T-junction-connected dead-end leg at NPPs. Unstable T/H response in the dead-end leg flows is a source of thermal fluctuation that can provoke thermal fatigue in auxiliary lines. These flows may contribute to the occurrence of stress corrosion too. This work will lead to a better understanding of the phenomenology and improved anticipation of the risk of auxiliary line failure. It is also hoped that this work can contribute to better control the flow and hence mitigate the risk of failure.

Past experiments available on this type of flows had insufficiencies for the validation of CFD methodologies. Currently, semi-empirical tools based on scaled-down experiments conducted at room temperature are used for the screening of potential problems, and plant measurements are used to monitor potential risks. The current prediction and screening methods have large uncertainties and can lead to wrong conclusions. Monitoring is also uncertain, time consuming and expensive.

A new benchmark proposal will be discussed during the next TG meeting in June 2024.

***III.C. Workshop (WS)***

The ninth **WS** of **OECD/NEA CSNI CFD for NRS (CFD4NRS-9)** [12] was held in February 2023, being hosted by the Texas A&M University at the Center for Advanced Small Modular and Microreactors (CASMR) in College Station, Texas.

The **CFD4NRS-9** WS brought together nuclear safety research organizations, TSOs, industry and academia, to exchange and share experiences and research results of the development, assessment, and applications of single-phase and multi-phase simulation tools acting at the small scale for NRS issues, and to address both CFD applications and the related experimental and instrumentation issues.

The workshop featured 3 Keynote Lectures, 2 Panel Sessions and 50 oral presentations. Sixty participants attended the workshop and the emphasis was on offering exposure to state-of-the-art of CFD applications and to move forward in the use of CFD within the industry and regulatory authorities.

Some key recommendations emerged from the WS. One of those is the necessity of “minimum standards” in contrast to the BPGs for CFDs to be used by the industry and regulatory bodies. Currently, the BPGs are too complex and resource demanding to become a staple in nuclear industry. To create confidence in a CFD result, other simpler methods should be developed. This was in line with another insight which is the common effort required to push for open source codes outside academia and research institutions. There are some benefits associated with using open source codes, but a common collaboration is deemed necessary for this transition to happen.

Finally, the WS was exposed to discussion and, to a certain degree, disagreement on the adequacy of numerical models, schemes and closure equations to model specific phenomena with different fluids or conditions, proving that experimental work coupled to numerical results, should still be part of the collaborative effort to expand the confidence in CFDs.

IV. Future WGAMA effort

***IV.A. Update and extension of CCVM (CSNI Code Validation Matrix)***

The WGAMA has made a significant effort to contribute and even promote reliable reactor safety assessment by providing code validation methods (e.g., CCVMs) and experimental data for the validation, and by enhancing the understanding of various kinds of T/H phenomena that may appear during reactor accidents. The CCVM was formulated for the preparation and utilisation of well-validated T/H codes by summarising the knowledge in a form of PIRT on the characteristics (and availability) of experimental facilities in terms of accident phenomena, which thus required code capabilities, with priority.

The WGAMA has developed many CCVMs since 1987, from the era of PWG-2, not only for the accident phenomena that may appear in a reactor system during an accident, but also for in-vessel core degradation [15 - 23]. Related experimental data are given through the NEA Data Bank for the SET and IET data [24, 25].

The review and update of the past established CCVMs is under planning in response to the recent updating of models and correlations employed in the computer codes for safety analyses. Furthermore, the safety assessment of advanced reactors requires precise representation of all new features in the accident phenomena. The CCVMs may need further harmonization and updating in response to such requirements to firmly confirm the applicability of the computer codes to the safety assessment of the advanced reactor design of interest.

***IV.B. Knowledge transfer: THICKET***

The Seminar on Transfer of Competence, Knowledge and Experience Gained through CSNI Activities in T/H Field (THICKET), is aimed to disseminate the T/H knowledge, competence, and experience acquired through CSNI activities during the last three decades to newcomers in the nuclear sector. A series of seminars has thus been organized to transfer implicit, explicit and tacit knowledge and experience acquired by senior experts who have contributed substantially to current activities of the CSNI. The next seminar will be organized over the experience gained from the past seminars conducted at Saclay (2004, France), Pisa (2008, Italy), Paris (2012, France), and Budapest (2016, Hungary), each with over 30 participants. The seminar is extended to the WGAMA contributions in the field of SA.

The format of the seminar will make use of a combination of lectures from the past four seminars, with emphasis on nuclear safety in T/Hs and SA area on key activities such as ISPs, CCVMs, SOARs, SEG reports, TOPs, code V&V, user effect, scaling, uncertainties, source term, fuel degradation, melt behaviors including MCCI and FCI, and long term management. The emphasis will be given to recent and priority activities including WGAMA-relevant NEA JPs. Collaborative effort with CSNI WGFS is foreseen to deepen understanding on fuel behavior under reactor accidents including latest-design fuels. Application of safety-relevant findings to the design of non-water cooled reactors will also be addressed.

V. NEA Joint Safety Research Projects (JPs) [26]

One of the major achievements of the NEA is the knowledge it has helped to generate through the organization of joint international research projects (JPs). Such JPs, primarily in the area of nuclear safety, enable interested countries, on a cost-sharing basis, to pursue research or the sharing of data with respect to areas or issues of high nuclear safety relevance.

The primary motivation for the JPs is to support the safe operation of nuclear installations. Typical safety research projects include:

* The conduct of sets of experiments that address safety knowledge gaps;
* The creation and maintenance of shared databases of operating experience that can be used to identify areas for enhancement and good practices; or
* The exchange of experimental information and modelling results to better understand safety-related phenomenology.

These JPs, while distinct from the program of work executed through the NEA Standing Technical Committees, have been of particular interest to the members of the CSNI, and have complemented the CSNI’s program of safety research activities. The JP principles and benefits are described in a video in the public webpage **[27]** and in a brochure **[28]**.

JPs contribute further to the development and preservation of technical capabilities. Together with experts, students and/or young professionals can be involved and benefited from sharing knowledge and different approaches. Furthermore, participants are encouraged to use staff visits and assignments for the transfer and further development of knowledge and capability.

In the area of T/H there are a series of JPs ongoing, which are briefly described below as example.

JPs benefits and perspectives were reviewed at a large event in January 2023 organized by NEA [29]. In the T/H area, it has been said it would be beneficial to establish for the future a more integrated approach using synergies and complementarities between available T/H facilities to address efficiently pending safety issues in current and future reactors, including SMRs. This is of particular importance in a period with the foreseen closure of key facilities and the development of new facilities addressing new designs, featuring passive safety systems.

**ETHARINUS [30]**, Experimental Thermal Hydraulics for Analysis, Research and Innovations in NUclear Safety (ETHARINUS) Project utilizes mostly the PKL facility at FRAMATOME in Germany. This project investigates T/H phenomena where the knowledge base for safety assessment is not sufficiently developed and provides qualified data for the development and validation of T/H codes and models used for the analysis of key safety issues. The project scope includes development of a common database to contribute to the experimental verification of cool down procedures, operation modes and systems (including passive ones) for different transients. An enhanced focus of the investigations is on studies related to loss of coolant accident (LOCA) addressing DEC for small break LOCA (SB-LOCA) scenarios and two-phase flow thermal-hydraulic phenomena for Intermediate break LOCA (IB-LOCA) scenarios, on studies related to core cooling performance under partial core blockage, on studies related to effectiveness of passive heat removal systems and on studies related to cool down procedures after a multiple steam generator tube rupture (MSGTR) scenario. ETHARINUS experiments are highly relevant to the improvement and validation of T/H safety codes and their use, as well as to the maintenance of competence and expertise in this field.

While the PKL facility will be dismantled by March 2024, the gained PKL data will be used for a new ISP addressing MSGTR related to DEC scenario that may occur in the hypothetical scenario of a combined msteam line break (MSLB) and MSGTR. The project participants are working in preparing a new proposal to ensure continued contribution to the T/H safety issues with alternate T/H facilities.

**ATLAS-3 [31]**, Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) Phase 3 Project utilizes the ATLAS test facility at KAERI. This project is referenced in the report "The Fukushima Daiichi (1F) NPP Accident: OECD/NEA Nuclear Safety Response and Lessons Learnt" **[32]** as one of the three new JPs based on existing research facilities that address safety issues related to the 1F accident.

This project is aimed to investigate the T/H field of accident scenarios of high safety relevance for both existing and future NPPs. Such investigations are of specific importance because they contribute to the validation of computer codes that are required in safety evaluation of light water reactors (LWRs) in order to simulate plant behavior during DBAs and DECs. They cover complex multi-dimensional single-phase and two-phase flow conditions.

**RBHT [33],** Rod Bundle Heat Transfer (RBHT) Project utilizes the RBHT test facility at Pennsylvania State University under the United States Nuclear Regulatory Commission (NRC) co-ordination. Modelling reactor core behavior under accident conditions with delayed re-introduction of cooling water (commonly referred to as reflood) is a challenge for safety analysis computer codes. Reflood T/H (e.g., post-critical heat flux (CHF) flow and heat transfer, (liquid droplet) entrainment, quench) remains a major contributor to code uncertainties in the simulation of accident and must be more deeply understood to enhance nuclear safety. As nuclear industry evolves, there is a need for additional data for power up-rates and new reactor designs.

Reflood heat transfer and rod bundle T/H have been studied since 1960s when concerns were raised on the effectiveness of Emergency Core Cooling System (ECCS) of NPP. Reflood T/H has been extensively studied and significant improvements were made in modelling and simulation. With the recent needs to improve further plant operating conditions, efforts to develop more “mechanistic” models for reflood T/H have been undertaken; that is, models for physical processes would be based on the fundamental mechanisms that govern T/H rather than be based on empirical correlations that were often restricted to a specific range of applicability. While existing reflood experiments had demonstrated the effectiveness of the ECCS and provided information suitable for the development and licensing of evaluation models, a reflood facility with advanced and detailed measurement capabilities should take an important next step.

Data from the RBHT test facility is necessary for evaluation of applicant licensing submittals, and for development and assessment of the NRC confirmatory analysis code, and they are of interest to industry, regulatory bodies, TSO and research organizations. The RBHT test facility was designed and constructed in 1998. Research has continued since then, and a wide variety of tests have been conducted to date.

**POLCA**, POol during Loss of Cooling Accident **[34]** is under development. The ASPIC and MIDI facilities at IRSN in Cadarache, France would be used. A limited number of tests have been performed in the time frame of the project [35, 36]. The POLCA project would pursue the experimental analysis of the behavior of spent fuel pools in LOCA conditions.

The objective of a potential future experimental program is:

* to enhance knowledge on accidental SFP and to provide thermal hydraulics data;
* to support the thermo-hydraulics model development and validation for SFP under loss of cooling accident;
* to evaluate some mitigation strategies concerning assembly management.

The POLCA project is proposed to focus on experiments at the MIDI and ASPIC large-scale facilities with dedicated instrumentation on thermohydraulic behavior of SFP during a LOCA. The project will contribute to the enrichment of an experimental database, useful to improve thermohydraulic numerical tools in pool conditions.

VI. Conclusions

International co-operation is the key for ever-enhanced global nuclear safety. In this regard, the WGAMA as one of the NEA/CSNI eleven WGs continues to advance its scientific and technological knowledge base needed for the prevention, mitigation and management of potential accidents in NPPs, and to facilitate international convergence on safety issues and strategies for the analyses and management of accidents. The WGAMA efforts, experiences and achievements for the safety assessment of operating NPPs including SA aspects will be of great help for the continuous safety improvements required for the advanced reactors including SMRs.

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Acronyms

AI artificial intelligence

AM accident management

ATLAS Advanced Thermal-hydraulic Test Loop for Accident Simulation

BDBA beyond design basis accident

BEPU best estimate plus uncertainty

CCVM CSNI Computer Validation Matrix

CFD computational fluid dynamics

CHF critical heat flux

CSNI Committee on the Safety of Nuclear Installations

DBA design basis accident

DEC design extension condition

DNS direct numerical simulation

ECCS emergency core cooling system

IET integral-effect tests

ISP international standard problem

JP joint project

LOCA loss-of-coolant accident

LWR light water reactors

MSGTR multiple steam generator tube rupture

ML machine learning

MSLB main steam line break

NEA Nuclear Energy Agency

NPP nuclear power plant

NRC Nuclear Regulatory Commission of USA

NRS nuclear reactor safety

OECD Organization for Economic Co-operation and Development

PIRT phenomena identification and raking table

PKL Primärkreislauf-Versuchsanlage / large-scale test facility of a PWR primary loop

POLCA POol during Loss of Cooling Accident

PSA probabilistic safety assessment

PWR pressurised water reactor

RBHT Bod Bundle Heat Transfer

SEG senior expert group

SFP spent fuel pool

SMR small modular reactors

SOAR state-of-the-art repost

TG task group

T/H thermal-hydraulics

THICKET Seminar on Transfer of Competence, Knowledge and Experience Gained through CSNI Activities in the Field of Thermal-hydraulics

TOP topical (or technical) opinion paper

TSO technical support organization

UQ uncertainty quantification

V&V verification and validation

WG working group

WS workshop

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