NEA
NUCLEAR ENERGY AGENCY

# ACCELERATING INTERNATIONAL COOPERATION ON SMR SAFETY RESEARCH

***The OECD/NEA Working Group on the Analysis and Management of Accidents (WGAMA)***


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*IAEA International Conference on Small Modular Reactors and their Applications, 21-25 October, 2024*

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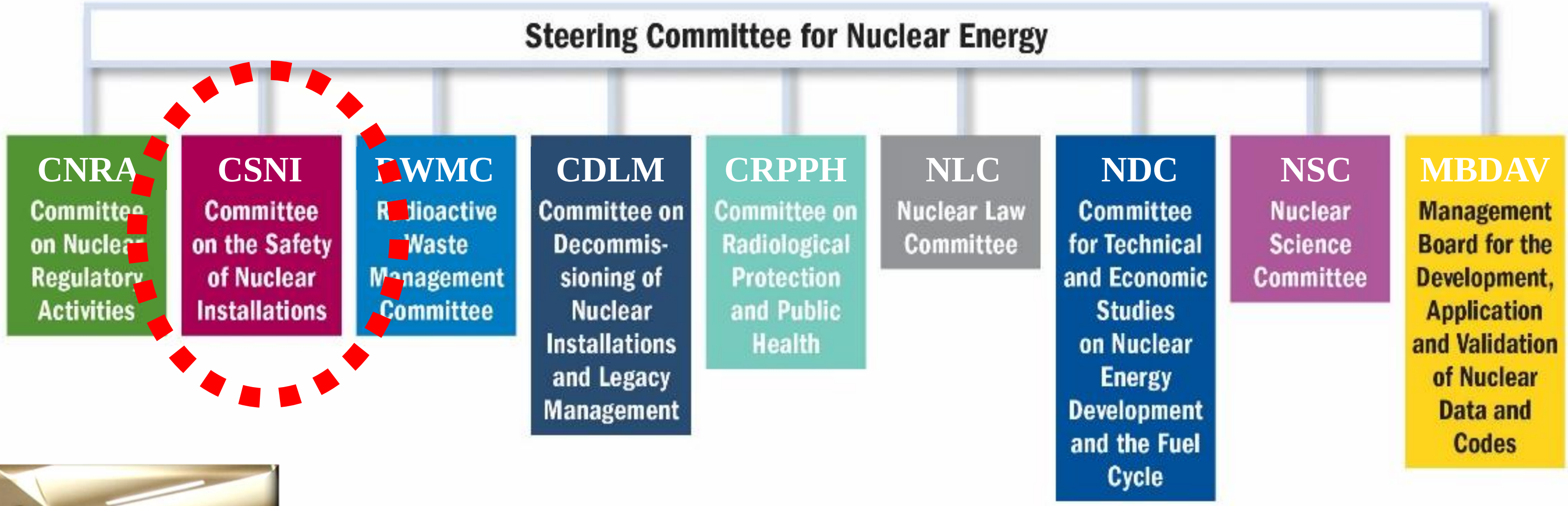
# The NEA

# 34 Countries Seeking Excellence in Nuclear Safety, Technology, and Policy

- ***Founded in 1958***





***8 Standing Technical Committees***

***1 Management Board -- NEA Databank***

***74 Working Parties and Expert Groups***

***26 Nuclear Safety Research Joint Projects***

# CSNI Overview

## Committee on the Safety of Nuclear Installations ( CSNI )

- CSNI Programme Review Group (PRG)
- Working Group on Risk Assessment (WGRISK)
- Working Group on Analysis and Management of Accidents ( WGAMA )
- Working Group on Integrity and Ageing of Components and Structures (WGIAGE)
  - Subgroup on the Integrity of Metal Components and Structures
  - Subgroup on the Ageing of Concrete Structures
  - Subgroup on Seismic Engineering
- Senior Expert Group on Preservation of Key Experimental Datasets (SEGPD)
- Working Group on Human and Organizational Factors (WGHOF)
- Working Group on Fuel Safety (WGFS)
- Working Group on Fuel Cycle Safety (WGFCS)
- Working Group on External Events (WGEV)
- Working Group on Electrical Power Systems (WGELEC)
- Expert Group on Small Modular Reactors (EGSMR)
- Expert Group on Fire (EGFR)

members
[25 Countries]

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# WGAMA Objective and Main Efforts

## *Objective*

The WGAMA assesses and, where necessary, strengthens the technical basis needed for the prevention, mitigation and management of potential accidents in nuclear reactors and related technologies, and to facilitate international convergence on safety issues, safety assessments and accident management (AM) measures and strategies.

## *Main Efforts*

- Exchanges technical experience and information relevant for resolving current or emerging safety issues;
- Promotes the development of phenomena-based models and computer codes used for the safety analysis, including the benchmarking exercises (*e.g., ISP (International Standard Problem)*);
- Assesses the state of knowledge in the areas relevant for the safety analysis;
- Where needed, promotes research activities aimed to improve such understanding, while supporting the maintenance of expertise and infrastructure in nuclear safety research.
  - ➢ *Supports the NEA Joint Projects (JPs) in the nuclear safety technical aspects.*

# WGAMA *Advances the understanding of accident phenomenology and enhances accident management (AM)*

Existing Reactors

### Technical reports
CFD supporting an enlarged use, Best Practice Guideline (**BPG**), **DEC-A**, Status Report **IVMR**, SOAR **H2**

### Code Benchmarks
**CFD** T-junction, **3 ISPs** [ACME, PKL, COAL], **Uncertainty** Quantification (**IUQ**) ATRIUM, Passive System (**PS**) PERSEO, FSI

### Code Validation, Data Preservation, and Knowledge Transfer
**CCVM**: CSNI Code Validation Matrix, **THEMPo: Meta-Data** preservation, collection, and qualification, **THICKET** seminars

### Technical meetings
**Specialist Workshops – BEPU-2024, SWINTH-2024, CFD4NRS-9 & -10**

### Opportunities
NEA Safety Research **Joint Projects** ATLAS3, ETHARINUS, RBHT, THEMIS, PANDA, ROSAU, FACE

### Co-ordination and Collaboration
NEA, IAEA, EC, ETSON …

### CSNI Status Report & Priorities
**Advances** in the analysis and management of accidents and future **challenges**

**2023**

**Thermal-Hydraulics, CFD, Severe Accident**

Advanced Designs

**Accident and Phenomena** → **Experiments and Data** → **Modelling** → **Computer Codes**

# WGAMA Thermal-Hydraulics (T/H) 1/2

## *Reliability assessment of Passive Systems (PSs)*

- **Status report** on thermal-hydraulic passive systems design and safety assessment
  - NEA/CSNI/R(2021)2 -- published this year
- **Main target** is
  to evaluate thermal-hydraulics (T/H) of the Passive Systems (PS) adopted or planned to be adopted into operating Nuclear Power Plants (NPPs), and into reactors with advanced design or undergoing licensing process including **SMRs**.
- **Key conclusions**
  - Suitable analyses and specific demonstrations of the adequacy of the design and operation of T/H PSs, namely those based on natural circulation, are required to demonstrate the safety of nuclear reactors relying on such systems.
  - PSs may contribute to improve the NPP safety,
    provided the safety targets are duly achieved by the methods, approaches and data (e.g., experimental database) available to industry and regulators.

# WGAMA Thermal-Hydraulics (T/H) 2/2

## *Modelling robustness of safety analysis codes*

- **Best-estimate (BE) computer codes** quantitatively evaluate **safety margin** by representing accident sequence/phenomena realistically.
- **Modelling robustness** of **BE codes**, with quantitative **uncertainty evaluation**
  - ➢ Central important subject to confirm the reactor safety design through **safety assessment**, also for emerging reactor designs including **SMRs**.
  - ➢ **Code validation** via **ISP*** exercises as a benchmark study to clarify (a) characteristics of each BE code, (b) uncertainty, and (c) user effects.
  - ➢ A series effort of **BE code uncertainty quantification:** BEMUSE, PREMIUM, SAPIUM, and ATRIUM -- [PWR DBA safety assessment]
- **Severe accident (SA):** Fukushima-Daiichi (1F) accident
  - ➢ **BSAF** benchmark to clarify the capability of SA analyses codes.
- **CFD:** 6 benchmarks in 10 years, including T-junction and GEMIX.

**Specialist WS**

**BEPU-2024**

- ✓ **Current status of development and use of BEPU** methods**
- ✓ **Challenges to apply it to multi-discipline (incl. multi-physics) modelling and simulation methods.**

***ISP: International Standard Problem**
****BEPU: Best Estimate plus Uncertainty**

**Forming <u>a basis of the safety evaluation techniques</u> for <u>emerging reactor designs including SMRs</u>.**

# WGAMA Computational fluid dynamics (CFD)

## *Modelling innovation in CFD*

- CFD solves a more mechanistic set of equations than safety analysis codes. However, …
  - **Turbulence** of single-phase flows still has difficulties to describe with sufficient fidelity.
  - Equations solved for CFD contain models mostly based on dedicated experimental results, namely **CFD-grade data**
- **Uncertainty quantification (UQ)** in the CFD nuclear safety assessment
  - Identification of uncertain parameters in case of "input uncertainties" problem is not straightforward.
  - Reduction of errors and computational cost are necessary to duly conduct the safety assessment. **Uncertainty reduction** mainly depends on the **quality of experimental data** in validation process.
- Request for **data and database preservation**
  - **A large volumes of data** is necessary even to confirm safe operation of nuclear installations.
  - **Databases specific to nuclear safety** are necessary for the V&V of CFDs
- **Water-cooled SMRs** are particularly of interest for potential CFD applications.
  - **Small length scale** and **small power level** may significantly reduce the computational power required for CFD analyses.
  - A challenge exists in turbulence models at high Rayleigh number values typically in large volumes

# WGAMA Severe Accident (SA) 1/3

## *Severe accident management (SAM) countermeasures*

- **Improving the SAM** by integrating the knowledge from recent research projects
  - **Synthesis report** "2018 International Severe Accident Management Conference (**ISAMC-2018**)" [NEA/CSNI/R(2021)9] provides various recent practices related to severe accident management guidelines (SAMGs), including expert judgment, simulators, field training tabletop exercises, and emergency drills.
- **Long-term management (LTM)** and **response to SA**
  - **Status Report** "Long Term Management and Actions for a Severe Accident in a Nuclear Power Plant" [NEA/CSNI/R(2018)13] provides recommendations for improving the LTM, based on compiled feedback from the past major SAs (TMI-2/1979, Chernobyl/1986, and Fukushima-Daiichi (1F)/2011).
  - **On-line specialist WS** "Reactor core and containment cooling systems: Long-term management and reliability (**RCCS-2021**)," [NEA/CSNI/R(2022)11] identified remaining knowledge gaps in this field. The WS emphasized the need to make further progress in the development of technical basis for demonstrating <u>the reliability of equipment under long-term SA conditions</u>, for both of existing and new reactor designs including **SMRs**.

# WGAMA Severe Accident (SA) 2/3

## *Advanced measurement methods and instrumentation*

- Severe accident management (**SAM**) relies on the measurement instrumentation that work properly as designed under harsh SA conditions.

- **SAMMI-2020 Specialist WS in 2020** [NEA/CSNI/R(2022)3] summarized the advanced measurement methods and instrumentation dedicated for SAM.
  - **Six topics:** Measurement methods for SAM, innovative instruments, best utilization of installed instrumentation with revised strategies, SA environment characterization and instrumentation qualification, and the LTM of SA-experienced NPPs such as TMI, Chernobyl and 1F.
  - **Keynote lecture** on **NuScale SMR** to consider role of measurement during postulated accident.

- **SWINTH-2024** was held, as an **updated specialist WS**, dealing with advanced measurement methods and instrumentation for …
  - Safety research experiments for both T/H and SA fields
  - Utilization for SAM in operating reactors

# WGAMA Severe Accident (SA) 3/3

## *State-of-Art-Report (SOAR) on hydrogen and carbon monoxide risk in late phases of SA*

- An overview of the main results obtained in managing the risks associated with hydrogen ($H_2$) and carbon monoxide (CO) in NPPs.
- **The SOAR** examines the behaviour of these combustible gases, starting with their production, distribution in the containment atmosphere, taking into account the effect of safety systems such as water sprays and recombiners (PAR), and ending with an assessment of their combustion regimes.
    - Each stage is closely linked, enabling risk mitigation measures and the management of serious accidents to be continuously refined.
- The SOAR, not only essential for **comprehensive containment analyses**, but also extends its applicability to **safety and maintenance considerations**, ensuring **a holistic approach** to the mitigation and management of **combustible gas risks under SA conditions**.

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WGAMA

# Update and extension of CCVM (CSNI Code Validation Matrix)



**Many CCVMs** have been developed ***since 1987*** to provide **a firm database** for **the validation of safety assessment codes for NPPs**.

- CCVM was formulated to prepare and utilize **well-validated T/H codes** by summarising the knowledge in a form of **PIRT** (Phenomena Identification Ranking Table) on the characteristics (and availability) of **experimental facilities** vs. **accident phenomena** that dominate different types of accidents and transients, which thus required code (modelling) capabilities, with priority.
  - **Safety evaluation of nuclear power plants by safety analysis** is an essential process to confirm the design of the reactor sufficiently safe.
- **Best set of available test data** has been successfully collected following the **code validation matrix** for assessment and improvement of codes, including quantitative assessment of uncertainties.
  - (1) accident phenomena in a reactor system, (2) in-vessel core degradation, (3) accident phenomena in a containment vessel, regarding **PWRs** and **BWRs**, and later (4) for **VVER** reactor phenomena.
  - **NEA Data Bank** stores test data for **SETs** (separate effects tests) and **IETs** (integral effect tests) identified in the **PIRTs** of **current CCVMs**.

# WGAMA Update and extension of CCVM (CSNI Code Validation Matrix) 2/3

WGAMA

Severe Accident

| | |
|---|---|
| NEA/CSNI/R(2014)3 | **Containment** Code Validation Matrix |
| NEA/CSNI/R(2001)4 | 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* |
| NEA/CSNI/R(2000)21 | **In-Vessel Core Degradation** Code Validation Matrix. |
| NEA/CSNI/R(1996)17 | CSNI **Integral test facility validation matrix** for the assessment of thermal-hydraulic codes for LWR LOCA and transients, 1996. Also referenced as: OCDE/GD(97)12 |
| NEA/CSNI/R(1996)16 | Evaluation of the **separate effects tests** (set) validation matrix, 1997. |
| NEA/CSNI/R(1995)21 | **In-vessel core degradation** code validation matrix, 1996. Also referenced as: OCDE/GD(96)14 |
| NEA/CSNI/R(1993)14 | **Separate effects test matrix** for thermal-hydraulic code validation, Vol.1 and 2, 1994. Also referenced as: OCDE/GD(94)82, 83 |
| NEA/CSNI-132 | **CSNI code validation matrix** of thermo-hydraulic codes for LWR LOCA and transients, **1987**. |

PWG-2

# WGAMA Update and extension of CCVM (CSNI Code Validation Matrix) 3/3

- **Review and update of the CCVMs** is necessary, because the safety assessment of advanced reactors requires precise representation of all new features in the accident phenomena.
  - CCVMs will need further harmonization and updating for such requirements by confirming the code applicability to the safety assessment of the advanced reactor design of interest.
  - Current safety analysis codes may be applicable to the advanced reactors including SMRs, as resulted from historical code development and validation efforts for the operating NPPs. However, the revised CCVMs will cover every aspect unique to accident sequence and phenomena at the advanced reactor designs.
  - CCVMs will be updated and extended to avoid any lacking in the safety assessments, based on the PIRTs with reactor prototype condition testing and/or the properly-scaled simulation experiments.
- **BE codes** validated with appropriate database designated by the **updated CCVMs** may fully cover expected accident conditions at the rector design of interest.
- **THEMPo**, WGAMA Task Group: “Harmonization of Methodologies for System Thermal-Hydraulics Experimental Meta-Data Preservation, Collection and Qualification”
  - In parallel, **THEMPo** TG develops, improves and harmonizes existing methodologies for collection, preservation, qualification, organization and use of an exhaustive set of experimental information (SEI), and provides a guide to create a relational database of the experimental meta data.

# WGAMA Knowledge transfer: THICKET

***THICKET-5** planned at Lucca, Italy in 2025*

- A series of **THICKET Seminar** "the Transfer of Competence, Knowledge and Experience Gained through CSNI Activities in T/H Field" has been organized to disseminate the key knowledge acquired through **CSNI nuclear safety activities** during the last three decades **to newcomers into the nuclear sector**.
  - ➢ **WGAMA senior experts** have contributed substantially to the current CSNI activities
  - ➢ **4 previous THICKET seminars** all done successfully, each with over 30 participants: Saclay (2004, France), Pisa (2008, Italy), Paris (2012, France), and Budapest (2016, Hungary)
- **The next 5th THICKET-5** will newly cover subjects in safety-relevant fields of T/H, CFD, SA, fuel safety, PSA (Probabilistic Safety Assessment) and NEA JPs including 1F-related JPs.
  - ➢ The seminar will provide **the lectures with emphasis on key recent and prioritized activities** such as ISPs, CCVMs, SOARs, SEG reports, TOPs, code V&V, user effect, uncertainties, scaling, source term (ST), fuel degradation, melt behaviours including MCCI and FCI, and LTM.
  - ➢ **Collaboration with CSNI WGFS and WGRISK** will deepen understanding on fuel behaviour during accidents and risk perspectives on the reactor accident consideration and decision making.
  - ➢ **Lecture content is utterly revised** with the fundamental yet precisely updated knowledge <u>to form a firm basis for each participant confronting any safety-relevant subjects</u> for operating and emerging reactor designs including water-cooled SMRs.

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# **WGAMA** Concluding Remarks

- **Whole WGAMA activities** and **recent major outcomes** reviewed
- **Technically meaningful achievements** identified in each of **T/H**, **CFD** and **SA** fields
  - ✓ Most of the achievements are applicable to the advanced reactors, although …
    T/H response during accidents may be a little or greatly different from those for the operating NPPs.
- **Future challenges** identified to properly respond for
  ***imminent safety assessment needs*** against emerging reactor designs including SMRs
  - ✓ **Safety assessment with safety analysis codes validated with appropriate database** is the key to assure safe operation and validity of counter-measures to accidents at NPPs.
- **Updating of CCVMs** will assure the code validation with a large amount experimental data which may appropriately cover reactor accident responses specific to new reactor designs.
- **THICKET**, the knowledge transfer seminar, will foster the knowledge base building necessary for the future experts to pursue the safety of nuclear reactors including SMRs.

➢ **The WGAMA efforts, experiences and achievements** for the **safety assessment** of **operating NPPs** will be of great help for **confirmation and continuous improvements of safety**, which are essential for the **emerging reactor designs including SMRs.**

# Thank you for your attention!

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