AN OVERVIEW OF PAST AND PRESENT ACTIVITIES IN THE SEVERE ACCIDENT DOMAIN WITHIN THE FRAMEWORK OF WGAMA

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Abstract

The objectives of the NEA-CSNI's Working Group on the Analysis and Management of Accidents (WGAMA) are to assess and where necessary strengthen the technical basis needed for the prevention, mitigation, and management of potential accidents in nuclear power plants, and to facilitate international convergence on safety issues and accident management analyses and strategies. In order to fulfill this objective, the working group undertakes:

- Exchange technical experience and information relevant for resolving current or emerging safety issues,
- Promote the development of phenomena-based models and codes used for the safety analysis, including the performance of benchmarking exercises,
- Assess the state of knowledge in areas relevant for the accident analysis and where needed,
- Promote research activities aimed to improve such understanding, while supporting the maintenance of expertise and infrastructure in nuclear safety research.

Continuing to be active in the severe accident field as the successor of the previous principle working group 4 (PWG4) of CSNI, WGAMA has accumulated an immense consolidated knowledge, which has been created along the years and reflects the improved understanding in very complex severe accident phenomenology, their modeling and their risk and safety relevance.

WGAMA activities related to severe accidents include exclusively the following technical areas: progression of accident into core damage and associated in-vessel phenomena; coolability of over-heated cores; ex-vessel corium interaction with concrete and coolant; in-containment combustible gas control; physical-chemical behavior of radioactive species in the containment. The activities mainly focus on existing reactors, but will also comprise applications for some advanced reactor designs.

Being established in 2000, WGAMA carried out the activities which had been initiated by the former PWG 4 group for the first few years, and initiated several more afterwards. A more targeted approach was introduced in 2005 to set-up the priorities and timeline of new activities. As a result, selection of the currently running activities was established by a careful review of all relevant severe accident issues and considering the status of research being conducted in 2006-2007. Several severe accident issues that have potential to be tackled by means of the standard OECD tools were identified and grouped under 11 main titles. The appropriate activity type was assigned to each issue under the main title, their risk and regulatory relevance was determined based on responses of WGAMA members and possible activities in the short, mid and long term were defined. Currently (2010-2011), the screening process is repeated since many of the international severe accident projects have been completed or are well advanced.

The present paper will provide a brief overview of the WGAMA activities carried out in the last three decades in the severe accident field. It will highlight the results achieved in fission product release, transport, deposition and retention behavior, in-vessel behavior of degraded cores and in-vessel protection, containment behavior and containment protection, as well as in reaching a common understanding of severe accident management measures and their treatment in risk assessment.

Keywords: CFD, OECD/NEA, Severe Accidents, Assessment, fission products, aerosols, coolability, retention, Benchmark exercise.

1. WGAMA's mandate, objectives and vision

Mandate

The Working Group on the Analysis and Management of Accidents (WGAMA) is responsible for activities in the area of severe accident domain. It covers a large field involving severe core damage to potential release of activity into the environment, specifically; in-vessel core degradation, coolability of over-heated cores and many different related phenomena; vessel failure and subsequent containment issues including; ex-vessel corium interaction with concrete and coolant; in-containment combustible gas distribution and control; physical-chemical behavior of radioactive species in the primary coolant system and containment. The activities currently focus on existing reactors, but will also comprise applications for some advanced reactor designs. Priority setting in activity selections is based on established CSNI criteria, in particular on safety significance as well as risk and uncertainty considerations.

Objective

The WGAMA objective is to assess and where necessary, strengthening of the technical basis needed to prevent, mitigate, and manage potential accidents in nuclear power plants, and to facilitate international convergence on safety issues and accident management analyses and strategies.

In order to fulfill this objective, the working group:

- Exchanges technical experience and information relevant for resolving current or emerging safety issues,
- Promotes the development of phenomena-based models and codes used for the safety analysis, including the performance of benchmarking exercises,
- Assesses the state of knowledge in areas relevant for the accident analysis and where needed,
- Promotes research activities aimed at improving such understanding, while supporting the maintenance of expertise and infrastructure in nuclear safety research.

The activities are normally carried out by small task groups, each is set up for performing a specific task under the WGAMA supervision.

The Working Group supports higher level working groups, CSNI and/or CNRA, and co-ordinates its work with other CSNI working groups, e.g., with WGRISK for risk related issues. All the activities follow the recommendations in the CSNI Operating Plan and are subject to endorsement of Program Review Group (PRG) and CSNI.

Vision

Assessment of the state of knowledge and progress regarding the understanding of phenomena and processes governing the occurrence, progression, and mitigation of potential accidents has been the continued vision of the working group. Demonstration of the performance and use of safety analysis codes is one of the main means for the assessment process. The second means is to convey the acquired knowledge regarding what is known and what is not known to the OECD community for their own use of safety assessment and other purposes in form of state of art reports, and other technical documents.

2. Brief Account of past and current WGAMA activities in the Severe Accident Domain

WGAMA traditionally has carried out three main types of the activities:

a) Establishing fundamental understanding on specific severe accident issues. These are specifically:

- Core degradation and behavior of molten corium during the early and late phase of the accident progression,
- Aerosol behavior,
- Iodine chemistry and behavior,
- Hydrogen generation and behavior,
- Fuel coolant interaction,
- Molten corium concrete interactions and
- Direct containment heating.

b) The second type of activities related with the assessment of knowledge gained in cross-cutting issues involving more then one phenomena taking place in space and time, namely:

- Fission product release and transport,
- Assessment of actual accident source terms,
- Accident management aspects of fission products,
- Accident management aspects of in and ex-vessel coolability and
- Impact of Short-Term Severe Accident Management Actions in a Long-Term Perspective.

c) The third type is the exchange of information and experience gained in integration of all relevant knowledge in the accident management, specifically:

- Implementation of severe accident management measures,
- Operator Training and Instrumentation Capabilities.

The WGAMA utilizes the following standard OECD tools to reach the targeted objectives for the above main group of activities:

- Preparation of collective opinion papers, and status reports,
- Launching international standard problems to aid benchmarking codes,
- Following the results of the related OECD projects that run outside of the WGAMA's mandates,
- Organizing specialist meeting/workshops to promote exchange of new information developed in individual member countries among their specialists and
- Launching preparation of state of art reports on particular areas where the accumulated knowledge in the last years can be distilled and consolidated.

The following chapters and subchapters include tables presenting all severe accident related activities and related publications since late 1980, which are available through http://www.nea.fr/html/nsd/docs/indexcsni.html.

2. Fundamental understanding on specific severe accident issues

The following subsections introduce individual activities and summarize the main outcomes. The order of introduction is not related with any relative significance of the individual phenomena.

2.1 Core degradation and behavior of molten corium during the early and late phase of the accident progression

Sustained lack of core cooling would eventually cause fuel to heat-up and not only to loose its integrity but to melt. The heat necessary to reach to the high temperature >1900 K is partially the decay heat, but most effective heating comes from the exothermal reaction due to steam and zircaloy cladding reaction, which also cause the generation of hydrogen. High temperature and later melting will also cause fission products trapped in the fuel matrix to be released into the coolant. The degradation and melting and further the relocation of molten material to lower elevations, perhaps freezing and followed by re-melting and hence its continued downward motion are complex processes and are additionally influenced by the operator actions with an attempt to stop the heating/melting with water injections. Additional complexity is the effect of control rod materials which would reach to melting at a much lower temperature then the fuel. Many different eutectic reactions and dissolution processes will bring the melting temperature of UO2 fuel from 3300 K down to about 2000 K. Many dedicated small and large scale experimental programs, accompanied by comprehensive separate effect tests were conducted to understand the processes and to provide data to model various individual phenomena and interactions among them.

Although not an experiment, TMI 2 accident provided rather limited data for developing understanding on what happened and also for assessing code models developed using other source of data. Data from various facilities at different scales, e.g., LOFT (USA), CORA (Germany), QUENCH (Germany), RASPLAV (Russian Federation), MASCA (Russian Federation), Phebus (France), were utilized in many international standard problems. The data from these facilities addressed the early phase core degradation and melting as well as the late phase molten pool behavior. Table 2.1 presents all the work of WGAMA and the former PWG4 as well as related OECD projects in the field since 1991.

As a matter of fact, the processes are too complex and also stochastic by nature and multidimensional regarding affecting parameters, the status of the knowledge has reached currently a level that predictions with models are rather satisfactory during the early phase of the core degradation, but deviating from what was measured later in the late phase drastically. However, it should also be noted that due to the strong technical difficulties it is impossible to measure all quantities in detail needed for the modelers.

Table 2.1: List of WGAMA and former PWG4 activities related to early and late phase core melt progressions

Title of the activity	CSNI/NEA report number and the year of publishing
Benchmark Exercise on an Alternative TMI-2 Accident Scenario	NEA/CSNI/R(2009)3, 2009
Main Results of the MASCA-1 and MASCA-2 Projects. Integrated application Report	NEA/CSNI/R(2007)15, 2007
OECD MASCA Project - Main result of the Phase 1 (2001-2004) - Integrated Report	NEA/CSNI/R(2004)23, 2004
ISP 46, PHEBUS FPT-1 Integral Experiment on Reactor Severe Accidents	NEA/CSNI/R(2004)18, 2004
ISP-45 (QUENCH 06 Test) - Final Comparison and Interpretation Report	NEA/CSNI/R(2002)2, 2002
OECD RASPLAV Project (1994-2000) - Behaviour of the Corium molten Pool under external cooling - Final Report	NEA/CSNI/R(2000)25, 2001
In-Vessel Core Degradation Code Validation Matrix	NEA/CSNI/R(2000)21, 2001
Status of Degraded Core Issues - Synthesis Paper, October 2000	NEA/CSNI/R(2001)5, 2001
ISP 36, CORA-W2 experiment on severe fuel damage for a Russian type PWR: comparison report	NEA/CSNI/R(1995)20, 1996
In-vessel core degradation code validation matrix	NEA/CSNI/R(1995)21, 1996
Summary of important results and SCDAP/RELAP5 analysis for OECD LOFT Experiment LP-FP-2	NEA/CSNI/R(1994)3, 1994
Summary and conclusions of the OECD Workshop on Large Molten Pool Heat Transfer, Grenoble, France	NEA/CSNI/R(1994)31, 1994

ISP 31, CORA-13 Experiment on Severe Fuel Damage	NEA/CSNI/R(1993)17, 1993
Phebus-SFD B9+ Experiment on the Degradation of a PWR type Core	NEA/CSNI/R(1992)17, 1992
In-vessel core degradation in LWR severe accidents: a state of the art report	NEA/CSNI/R(1991)12
TMI-2 Analysis exercise final report	NEA/CSNI/R(1991)8, 1992
TMI-2 examination results from the OECD-CSNI program	NEA/CSNI/R(1991)9, 1991
An Account of the OECD LOFT Project	NEA/CSNI-181/LOFT-T-3907, 1990

2.2 Aerosol behavior

Nuclear aerosols are generated from the condensing fission product vapors released from degraded or melting fuel or released during the core concrete interactions. Additionally non-active aerosols are generated form the condensing vaporized structural materials in the core region and from the vaporized concrete constituents. Once they are generated their transport and depletion processes depend on the aerosol physics as well as the thermal-hydraulic conditions which control the aerosol transport/removal processes. Table 2.2 provides all activities carried out by WGAMA and former PWG4 in the area. The first activity in this domain is dated back to 1980 and was a CSNI specialists meeting on nuclear aerosols in reactor safety (Gatlinburg, Tennessee, USA). The bulk of the aerosol related activities between 1990 and 2000 involved conduction of international standard problems and organizing workshops. The standard problems or code comparison exercises covered the assessment of the codes under well defined conditions and addressed specific aspects, e.g., hygroscopicity (using the data obtained in AHMED facility, Finland) or interactions with steam condensation and transport in multi compartment models of containment (such as those used the VANAM and KAEVER test results obtained in the Battelle test facility in Frankfurt, as used in ISP 37 and 44, respectively). These tests and code benchmarking indicated the role of the correct modeling of hygroscopic nature of the aerosol material and steam condensation driven aerosol growth on the depletion rate of the aerosols. The deposition and resuspension behavior of aerosols in a horizontal tube using one of the Storm tests was subject of ISP 40, and indicated the deficiencies in the modeling of rather complex nature of the resuspension phenomena, which still poses a challenge for the codes to capture.

Three workshops (1980, 1990 and 1998) provided forums where the advances in the experimental and modeling aspects of nuclear aerosols were presented and discussed. The last coordinated effort was the preparation of a state of art report on nuclear aerosols, which was issued in 2009. The report provides the current understanding of many different aspects of aerosols; the aerosol physics relevant to nuclear aerosols and droplets subject to accident environment, their transport and depletion processes, modeling of these processes, the capabilities of codes in predicting the aerosol behavior in reactor primary coolant system and containment and removal of aerosol particles by engineered safety systems, such by pressure suppression pools, containment venting and other filters, containment sprays.

Currently, international projects on specific futures of aerosol behavior and transport, e.g., Phebus FP (France), OECD-THAI (Germany), ARTIST (Switzerland), have generated invaluable data, but not yet available for OECD community for coordinated undertakings.

Table 2.2: List of all WGAMA and former PWG4 activities related to aerosol behavior.

Title of the activity	CSNI/NEA report number and the year of publishing
State-Of-the-Art Report on Nuclear Aerosols (NARSOAR)	NEA/CSNI/R(2009)5, 2009
ISP 44, KAEVER tests, Comparison and Interpretation Report	NEA/CSNI/R(2003)5, 2003
Specialist Meeting on Nuclear Aerosols in Reactor Safety, Köln Germany	NEA/CSNI/R(1998)4 and NEA/CSNI/R(1999)5, 1999

ISP 40, Aerosol Deposition and Resuspension in STORM Test SR 11	NEA/CSNI/R(99)4, 1999
3rd Specialist Meeting on Nuclear Aerosols in Reactor Safety, Cologne, Germany	NEA/CSNI/R(1998)4, 1998
ISP37, VANAM M3 - A Multi compartment aerosol depletion test with hygroscopic aerosol material	NEA/CSNI/R(1996)26, 1996
AHMED Code comparison exercise	NEA/CSNI/R(1995)23, 1995
Short overview on the definitions and significance of the late phase fission product aerosol/vapor source	NEA/CSNI/R(1994)30, 1994
Physical and chemical characteristics of aerosols in the containment	NEA/CSNI/R(1993)7, 1993
Effects of hydrogen combustion on fission products and aerosols	NEA/CSNI/R(1993)6, 1993
Workshop on aerosol behavior and thermal-hydraulics in the containment, Fontenay-aux-Roses, 1990	NEA/CSNI/R(1992)1, CSNI Report n° 176
Workshop on aerosol behavior and thermal-hydraulics in the containment: proceedings, Fontenay-aux-Roses, France	NEA/CSNI-176, 1990
CSNI Specialist Meeting on nuclear aerosols in reactor safety, Karlsruhe, Germany, 1984	NEA/CSNI-95, 1985
CSNI specialists meeting on nuclear aerosols in reactor safety, Gatlinburg, Tennessee, 1980	NEA/CSNI-45 NUREG/CR-724 ORNL/NUREG/TM-404, 1980

2.3 Iodine chemistry and behavior

Due to its toxicity if inhaled, the fission product iodine has received a large interest since many decades for its numerous different aspects, regarding its release from the degraded fuel, its chemistry determining its physical form and chemical speciation, and its transport and depletion. Substantial efforts have been spent to create understanding in all these many issues, which finally determine its amount and speciation in the containment and finally its amount into the environment. Iodine, with its many oxidation stages (-1 to +7), is very reactive element and undergoes, especially under radiation fields, many different reactions, particularly in aqueous phase, some of which are responsible for the generation of volatile species (e.g., elemental iodine, volatile organic iodides) if the conditions promote their generation. These reactions are complex to understand and model and may become even more complex once other reaction partners, such as organic residuals, paints, and other fission products or impurities, e.g., nitrous oxides, organic residuals, are present.

The first joint undertaking in the iodine area was the 1. Iodine workshop organized in 1985 in UK, followed by three more workshops, in 1989, 1996, and 1999, where, the results of dedicated experiments and models developed were presented. In addition to the modeling of iodine behavior within the context of the accident progression, accident management aspects of the iodine was subject of an OECD Workshop in 1999. Table 2.3 lists all the related activities.

The WGAMA and its former group organized three standard problems facilitating code to code and code to data comparisons utilizing experimental data from experiments (RTF (Canada), Caiman (France) and Phebus FPT1 (France)). The preparation of the state of art report on iodine chemistry in 2007 is the last joint undertaking and summarized the acquired knowledge. It pointed out that the iodine chemistry under clean laboratory conditions can be captured with the models as build in the special containment iodine codes with success. However, the report also pointed out to the weaknesses in the understanding and hence the lack of comprehensive modeling once the conditions and participating reaction partners have not been experimentally studied. Currently, iodine poses a continued challenge to the researchers and modelers, and hence dedicated experimental and modeling efforts are continuing in many international projects, e.g., OECD-BIP, International Source Term Project (ISTP), OECD-THAI, EU-SARNET 2. The recent experimental programs have generated numerous data, but are not yet available to OECD community for joint undertakings.

Table 2.3: List of all WGAMA and former PWG4 activities related to iodine behavior

Title of the activity	CSNI/NEA report number and the year of publishing
State of the Art Report on Iodine Chemistry,	NEA/CSNI/R(2007)1, 2007
ISP-41, ISP – 41 FU/2 Follow-up exercise (Phase II): Iodine Code Comparison Exercise against CAIMAN and RTF Experiments	NEA/CSNI/R(2004)16, 2004
ISP-46, PHEBUS FPT-1 Integral Experiment on Reactor Severe Accidents,	NEA/CSNI/R(2004)18, 2004
ISP-41, FU/1 Follow-up exercise: Containment Iodine Computer Code Exercise Parametric Studies	NEA/CSNI/R(2001)17, 2001
Insights into the Control of the Release of Iodine, Cesium, Strontium and other Fission Products in the Containment by Severe Accident Management,	NEA/CSNI/R(2000)9, 2000.
ISP 41, Containment Iodine Computer Code Exercise Based on Radioiodine Test Facility (RTF) Experiment,	NEA/CSNI/R(2000)6, 2000.
CSNI Workshop on Iodine in Severe Accident Management. Helsinki, Finland	NEA/CSNI/R(1999)7, 1999
The fourth CSNI Workshop on the chemistry of iodine in reactor safety, Würenlingen, Switzerland)	NEA/CSNI/R(1996)6, 1996
The third CSNI Workshop on Iodine Chemistry in Reactor Safety, Tokai-mura, Japan, 1991	NEA/CSNI/R(1992)5/JAERI-M 92-012, 1992 NEA/CSNI/R(1991)15
The second CSNI workshop on iodine chemistry in reactor safety, Toronto, Canada	NEA/CSNI-149/AECL-9923, 1989
The first CSNI specialists' workshop on iodine chemistry in reactor safety, Harwell, England	NEA/CSNI-114, 1985

2.4 Hydrogen generation and behavior

Many dedicated experiments have demonstrated that hydrogen if released in air or air – steam mixture and would reach a concentration of >7 %, can deflagrate. It may even detonate if much higher H_2 concentration levels are achieved. Hydrogen pollsters or layers with high H_2 concentrations >7 can occur under a variety of conditions in reactor containments or other buildings. The main source of the hydrogen is the oxidation of the fuel cladding with steam liberating hydrogen once the cladding (or other metals) reaches high enough temperature (>1100 K). The significant hydrogen generation starts at a zircaloy cladding temperature of about 1200 K and its generation rate becomes very excessive if the temperature exceeds 1500 K. Water addition to the already degraded core can cause much violent hydrogen generation since unexposed metallic surfaces may be exposed to steam due to shattering of the molten-solid debris. The other source of hydrogen generation is the core concrete interaction, during which un-oxidized zircaloy and iron bars in the concrete can be oxidized by the steam liberated from the decomposing concrete. The total H_2 released from the core-concrete interaction may be as high as, even higher then the amount generated in the reactor vessel. Radiolysis of sump water can cause a steady generation of hydrogen, and its amount is dependent on the amount of dose rate generated by the deposited activity in the sump water and its constituents. Core concrete interaction may also release substantial amounts of CO_2 depending on the concrete type. CO_2 may be reduced by H_2 to produce carbon monoxide, however the yield is dependent on the temperature and H_2 amount. CO is to be accounted as an additional increase in the flammable gas concentration.

Not only the understanding and modeling of the release of H_2 from the primary coolant system into the containment is important but also understanding conditions leading to development of a well mixed H_2 or stratified H_2 layer in the containment during the release phase and the related modeling are equally crucial. Hydrogen control, e.g., controlled burn of hydrogen or removal by recombination by the use of recombiners to avoid its potential deflagration becomes extremely important to secure structural integrity of containment or other reactor buildings due to spontaneous pressure build-up if hydrogen deflagration/detonation is not hindered. WGAMA/PWG4 has spent significant efforts (Table 2.4) to produce consensus views on in- and ex-vessel hydrogen generation. Further efforts were also given to produce consensus views on the flame acceleration and deflagration-to-detonation transition in 2000 based on advances made after the preparation and release of the state of the art report issued in 1992. Assessment of knowledge as modeled in codes to predict hydrogen behavior in two different facilities was achieved by conducting two ISPs. The state art report on Containment Thermalhydraulics and Hydrogen Distribution issued in 1999 produced sufficient guidance for further research, which were carried out later in three OECD experimental programs, THAI, SETH 1 and SETH 2.

Because of the consequences of possible hydrogen deflagration up to detonation on the structural integrity of the containment significant efforts were given to understand how to manage hydrogen with dedicated safety systems (recombiners, spark plugs, sprays). Workshops organized to foster exchange of practical experiences and methods for the mitigation among the specialists of the OECD countries. As this paper is prepared ISP 49 on modeling the hydrogen combustion using data from two tests carried out in the THAI and ENACCEF facilities is close to completion.

Table 2.4: List of all WGAMA and former PWG4 activities related hydrogen behavior

Title of the activity	CSNI/NEA report number and the year of publishing
OECD THAI Project Final Report on Hydrogen and Fission Product issues relevant for containment safety assessment under severe accident conditions	NEA/CSNI/R(2010)3, 2010
In-Vessel and Ex-Vessel Hydrogen Sources - Report by NEA Groups of Experts	NEA/CSNI/R(2001)15, 2001
Technical Notes on Ex-vessel Hydrogen Sources	NEA/CSNI/R(2000)19, 2000
Carbon Monoxide - Hydrogen Combustion Characteristics in Severe Accident Containment Conditions	NEA/CSNI/R(2000)10, 2000
State of Art Report on Flame Acceleration and Deflagration to Detonation Transition in Nuclear Safety	NEA/CSNI/R(2000)7, 2000
State-of-the-Art Report on Containment Thermalhydraulics and Hydrogen Distribution	NEA/CSNI/R(1999)16, 1999
Implementation of hydrogen mitigation techniques during severe accidents in nuclear power plants	NEA/CSNI/R(1996)27, 1996
The Implementation of hydrogen mitigation techniques: summary and conclusions: OECD Workshop, Winnipeg, Manitoba, Canada	NEA/CSNI/R(1996)9, 1996
CSNI Workshop on the implementation of hydrogen mitigation techniques, Winnipeg, Wanitoba	NEA/CSNI/R(1996)8, 1996
ISP35, NUPEC hydrogen mixing and distribution test (TEST M-7-1)	NEA/CSNI/R(1994)29, 1994
ISP 29, Distribution of hydrogen within the HDR containment under severe accident conditions	NEA/CSNI/R(1993)4, 1993
Hydrogen management techniques in containment,	NEA/CSNI/R(1993)2, 1993
A state-of-the-art report on flame acceleration and transition to detonation in hydrogen/air diluent mixtures	NEA/CSNI/R(1992)3, 1992

2.5 Fuel Coolant Interactions

When molten corium falls in water pool it may produce steam explosion under certain conditions and may yield to a release of very fast and enormous energy input which can not be absorbed readily by the medium, i.e., water pool and the gas space above. The results of the explosion may mean a large spontaneous increase in the pressure of the gas space as well as pressure pulsations affecting the surrounding structures. The cause for the steam explosion, so called triggering, after conduction of many experimental programs and development of models in the last, at least, two decades, is still not yet fully understood. Although, underlying conditions leading to the steam explosion has been narrowed, however, exact conditions causing the explosion is yet subject of current OECD project SERENA 2 and laboratory scale experiments in some European countries.

The issue has been tackled by WGAMA and its former group PWG4 as well as in the related OECD projects since 1994 (Table 2.5) in terms of organizing workshops, preparing a technical opinion paper as well as launching OECD projects SERENA 1 and 2, where specific data have been made available providing very useful information for the model development/assessments. SERENA 1 project highlighted with limited number of reactor calculations that the fuel coolant interaction, if it takes place in a reactor pressure vessel, will possibly not impair the structural capacity of the pressure vessel due to the large enough safety margins encompassing any possible underestimation of the code predictions, however, if it happens in the cavity of the containment, it may cause local structural damage.

Table 2.5: List of all WGAMA and former PWG4 activities related to Fuel Coolant Interactions

Title of the activity	CSNI/NEA report number and the year of publishing
OECD Research Programme SERENA on Fuel-coolant Interaction - Steam Explosion Resolution for Nuclear Applications	NEA/CSNI/R(2007)11, 2007
OECD Research Programme SERENA: Steam Explosion Resolution for Nuclear Applications: Phase 1: Identification of relevant conditions and experiments for fuel-coolant interactions in nuclear power plants	NEA/CSNI/R(2004)7, 2004
Technical Opinion Paper on Fuel-Coolant Interaction	NEA/CSNI/R(1999)24, 1999
CSNI Specialist Meeting on Fuel Coolant Interaction, JAERI-Tokai, Japan	NEA/CSNI/R(1997)30, 1997
Technical note on ex-vessel core melt debris coolability and steam explosions	NEA/CSNI/R(1996)24, 1996
CSNI Specialists Meeting on Fuel-Coolant Interactions, Santa Barbara, California, USA,	NEA/CSNI/R(1993)8, 1994
Steam explosions and reactor safety	NEA/CSNI-74, 1982

2.6 Molten corium concrete interaction

Activities regarding molten corium concrete interactions were given emphasis in the late 1980s and early 1990s as a result of many experimental programs conducted in this time frame. The programs focused on the extent of the downward and sideward erosion of either siliceous or carbonate base concrete employed in small scale test facilities and provided data on the erosion rates. In addition, stopping the interaction by water flooding at the top was studied. Certain tests also provided specific data on radionuclide release from the corium-concrete interaction involving special corium composition containing fission product simulants. The activities are listed in Table 2.6. Since 1994, although there is no more work launched by PWG4/WGAMA, further efforts have been spent in different countries. The last is a recently completed OECD project MCCI, carried out in USA, which provided valuable information for the modelers and assessment of the computer codes. Currently, preliminary discussions are initiated for preparation of a state of art report on MCCI.

Table 2.6: List of former PWG4 activities related to molten corium concrete interactions

Title of the activity	CSNI/NEA report number and the year of publishing
Overview of current approaches regarding the use of water to cool a molten core in the containment in ten OECD member countries	NEA/CSNI/R(1994)34, 1994
Summary and recommendations of the Specialist meeting on molten core debris-concrete interactions, Karlsruhe, Germany	NEA/CSNI/R(1993)5, 1993
ISP 30, BETA V5.1 Experiment on Melt-Concrete Interaction, Comparison Report	NEA/CSNI/R(92)19, 1992
Second CSNI Specialist Meeting on Molten Core Debris-Concrete Interactions, Karlsruhe, Germany	NEA/CSNI/R(1992)10, 1992
Benchmark exercise on the chemical modeling of the release of radionuclides due to core-concrete interactions	NEA/CSNI-164, 1989
ISP 24, SURC-4 experiment on core-concrete interactions	NEA/CSNI-155, 1988
Report of Task Group on Ex-Vessel Thermal-Hydraulics Corium/Concrete Interactions and Combustible Gas Distribution in Large Dry Containments	NEA/CSNI-143, 1987

2.7 Direct containment heating (DCH)

If the pressure vessel of a PWR fails at high pressure (up-to normal operating pressure), the molten corium accumulated in the lower plenum might be aerosolized when ejected outside of the reactor pressure vessel. If the small melt particles find their way into the containment air space, due to extensive high heat transfer from the gluing particles into the containment atmosphere, the pressure of the containment might increase spontaneously and eventually exceed the failure pressure of the containment. Such a situation could produce a catastrophic containment failure. Following experimental investigations in USA in late 1980's and early 1990's, efforts were given to consolidate the acquired knowledge, especially to assess the safety of operating plants. Table 2.7 presents the past activities regarding establishing the status and preparing a state of art report on the issue. The outcome of the tests and the state of the art report is that DCH is very reactor containment design specific issue, and the likelihood of reaching to high pressure melt ejection is rather low due to improvements in the plant hardware for a successful depressurization. Since then, there has been no more activity undertaken by WGAMA/PWG4. However, it should be noted that currently an experimental program in Karlsruhe institute of Technology in Germany is underway to experimentally study the issue in a scaled-down facility representative of a German PWR design.

Table 2.7: List of former PWG4 activities related to direct containment heating

Title of the activity	CSNI/NEA report number and the year of publishing
State-of-the-art report on High pressure melt ejection (HPME) and direct containment heating (DCH)	NEA/CSNI/R(1996)25, 1996
Report of task group on ex-vessel thermal-hydraulics: status of direct containment heating in CSNI member countries	NEA/CSNI-153, 1989

3. Activities related to the assessment of knowledge gained in cross-cutting issues involving more than one phenomena taking place in space and time

The activities introduced in the following sections involve usage of the knowledge and experience gained in the individual areas as introduce in the previous sections but bring several of them into a context involving interactions among them and taking place in space and time. Again, the order of introduction in this text does not follow any relative importance.

3.1 Fission product release and transport

Fission product release and transport is being driven by on one hand the core degradation process for the release and the thermal-hydraulic conditions for the transport, on the other hand by high temperature chemistry which determines the physical form i.e., gaseous or particulates and speciation of the fission products. The integral tests, e.g. LOFT FP2, Phebus FP, produced integral data, and demonstrated the importance of capturing the interplay among various phenomena and parameters, which are otherwise not possible to be generated in separate effect tests. An example to the importance of the integral tests is the effect of released control rod materials on the speciation of the iodine.

Most of the WGAMA/PWG4 works in the area were conducted in 1990's and helped in producing a state of art report in 1994 (Table 3.1). Continued efforts in 2000s focused on the prediction of the release and transport processes using the Phebus FPT1 data in the international standard problem 46. International projects, e.g., Phebus FP, International Source Term Project (ISTP), have currently provided a vast amount data and the accompanying analytical programs have generated deep understanding of underlying processes. It may be that depending on the availability of the information, future joint undertakings would update the state of the art as built in 1994.

Table 3.1: List of WGAMA and former PWG4 activities related to fission product release and transport

Title of the activity	CSNI/NEA report number and the year of publishing
OECD/NEA THAI Project Final Report - Hydrogen and Fission Product issues relevant for containment safety assessment under severe accident conditions	NEA/CSNI/R(2010)3, 2010
ISP 46, PHEBUS FPT-1 Integral Experiment on Reactor Severe Accidents	NEA/CSNI/R(2004)18, 2004
Insights into the Control of the Release of Iodine, Cesium, Strontium and other Fission Products in the Containment by Severe Accident Management	NEA/CSNI/R(2000)9, 2000
Current evaluation of the Chernobyl reactor accident release	NEA/CSNI/R(1996)2, 1996
The Chernobyl reactor accident source term: development of a consensus view	NEA/CSNI/R(1995)24, 1996
Short overview on the definitions and significance of the late phase fission product aerosol/vapor source	NEA/CSNI/R(1994)30, 1994
Specific features of cesium chemistry and physics affecting reactor accident source term predictions	NEA/CSNI/R(1994)28, 1994
ISP 34, Falcon code comparison	NEA/CSNI/R(1994)27, 1994
Primary system fission product release and transport: a state-of- the-art report	NEA/CSNI/R(1994)2, 1994
Effects of hydrogen combustion on fission products and aerosols	NEA/CSNI/R(1993)6, 1993
Source term uncertainties: recent developments in understanding fission product behavior	NEA/CSNI/R(1992)2, 1992
An Account of the OECD LOFT Project	NEA/CSNI-181/LOFT-T-3907, 1990

3.2 Assessment of actual accident source terms

In 1990's three activities were launched (Table 3.2). The first one was reviewing the national source term positions and practices in OECD countries. The next one was achieving consensus view on the specific features of cesium chemistry and physics affecting reactor accident source term predictions. The last activity performed in 1996 was to generate a consensus views on the source terms from the Chernobyl accident.

Table 3.2: List of former PWG4 activities related to assessment of accident source terms

Title of the activity	CSNI/NEA report number and the year of publishing
Current evaluation of the Chernobyl reactor accident release	NEA/CSNI/R(1996)2, 1996
Specific features of cesium chemistry and physics affecting reactor accident source term predictions	NEA/CSNI/R(1994)28, 1994
Current national source term positions and practices in OECD member countries, 1990.	NEA/CSNI/R(1991)3, 1991

3.3 Accident management aspects of fission products

The source term mitigation aspects of the accident management were the first activity undertaken in 1989. Accident management aspects of iodine in severe accident management was the focus of an OECD workshop organized in 1999, where, experimental and modeling information on many different aspects of severe accidents, such as effect of cable pyrolysis, iodine behavior under accident conditions, etc, were presented. The workshop highlighted the fact that in limited number of power plants active measures have been incorporated to control the sump pH to hinder the iodine volatilization in containment sump. The third PWG4 undertaking was to create insights into the control of release of iodine, cesium, strontium and other fission products in the containment by severe accident management and produced the results of a survey displaying what specific measures were taken in some responding OECD countries regarding iodine. Table 3.3 depicts the references of these activities.

Table 3.2: List of former PWG4 activities related to assessment of accident source terms

Title of the activity	CSNI/NEA report number and the year of publishing
Insights into the Control of the Release of Iodine, Cesium, Strontium and other Fission Products in the Containment by Severe Accident Management	NEA/CSNI/R(2000)9, 2000
CSNI Workshop on Iodine in Severe Accident Management. Helsinki, 1999	NEA/CSNI/R(1999)7, 1999
Source term mitigation aspects of accident management	NEA/CSNI-165, 1989

3.3 Accident management aspects of in and ex-vessel coolability

Table 3.3 presents the main activities regarding the in- and ex-vessel flooding regarding the coolability of the degraded core in the reactor pressure vessel and corium overlying on the pedestal. Preparation of a situation report on the in-vessel core debris coolability through external flooding of the reactor pressure vessel was the first activity carried out in early 1990s. Since then the melt retention in the reactor pressure vessel by external vessel cooling for some reactor designs up to about 1000 MWe is an experimentally and through model simulations validated measure provided that the designs allow the external flooding. Two workshops in the in-vessel coolability area were conducted, one in 1998 and the second and the last one in 2009, where papers and discussions focused on conditions favoring coolability of the degraded core in the pressure vessel. The possible adverse effect of in-vessel flooding of degraded core with inadequate flooding rate was subject to strong discussions in the last workshop held in 2009. Ex-vessel debris coolability was a subject of an OECD workshop held in 2000, which pointed out the conditions for potentially achieving coolable state.

Table 3.3: List of WGAMA/PWG4 activities related to Accident management aspects of in and ex-vessel flooding

Title of the activity	CSNI/NEA report number and the year of publishing
Joint OECD/CSNI and SARNET workshop on In-vessel coolability	NEA/CSNI/R(2010)11, 2010
OECD/CSNI Workshop on Ex-Vessel Debris Coolability - Summary and Recommendations, Karlsruhe, Germany	NEA/CSNI/R(2000)14, 2000
OECD RASPLAV Project (1994-2000) - Behaviour of the Corium molten Pool under external cooling	NEA/CSNI/R(2000)25, 2000
Degraded Core Quench : Summary of Progress 1996 -1999	NEA/CSNI/R(1999)23, 1999
OECD/CSNI Workshop on In-vessel Core Debris Retention and Coolability, Garching, Germany, 1998	NEA/CSNI/R(1998)21, 1998
In-Vessel core debris cooling through external flooding of the reactor pressure vessel: situation report	NEA/CSNI/R(1994)6, 1994
Core debris cooling with flooded vessel or core-catcher heat exchange coefficients under natural convection	NEA/CSNI/R(1994)32, 1994
Overview of current approaches regarding the use of water to cool a molten core in the containment in ten OECD member countries	NEA/CSNI/R(1994)34, 1994

3.4 Impact of Short-Term Severe Accident Management Actions in a Long-Term Perspective

Impact of short-term severe accident management actions in a long-term perspective was subject to an activity. It pointed out the effects of mitigation actions that might be effective in the short term but might produce some possible adverse effects in the long term. Table 3.4 displays the reference. The joint work addressed the use the new knowledge to improve the short- term severe accident management strategies to avoid or minimize the potential adverse effects in the long term. .

Table 3.4 WGAMA activity related to the Impact of Short-Term Severe Accident Management Actions in a Long-Term Perspective

Title of the activity	CSNI/NEA report number and the year of publishing
Impact of short-term severe accident management actions in a long-term perspective	NEA/CSNI/R(2000)8, 2000

4. Exchange of information and experience gained in integration of all relevant knowledge in the accident management

Management of a severe accident from the entry into a severe accident until stable plant conditions are reached produces many different challenges to the operators and crises centers and other relevant organizations since the cause to enter a severe accident might be a result of internal or external events triggering multiple and possibly concurrent failure of equipment. Additionally, since nuclear reactors are not equipped with diverse instruments like the experimental facilities do have, the detailed information for important events taking place in the reactor is rather limited. Potential failure of the instruments due to adverse conditions produces additional challenges to the operators to interpret the situation. Ability to make right decisions even though the assessment of what really happens could be extremely difficult adds another complexity and challenge to the managing the accident. After the TMI 2 accident severe accident management guidelines have been developed to help the operators to take the right actions. Since then, operators are continuously been trained, however, the level of the disaster caused by external or internal events might challenge many systems (availabilities and functionalities) as well as the ability of the operators coping with severe accident conditions. WGAMA and former PWG4 have given a large focus to establish forums in form of specialist meetings and workshops to exchange information and experience among all stake holders. Table 4.1 presents all related activities carried out since early 1990s.

Each workshop or specialist meetings focused on specific aspects of the accident management. As an example, the last one in 2009 emphasized the link of severe accident management to the probabilistic risk assessment, and the previous one in 2001 focused on the status of the implementation in various reactors operating in the member countries.

Table 4.1 WGAMA and former PWG4 activities related to the Accident Management

Title of the activity	CSNI/NEA report number and the year of publishing
OECD/CSNI Workshop on Implementation of Severe Accident Management Measures, Böttstein, Switzerland, 2009	NEA/CSNI/R(2010)10, 2010
OECD/CSNI Workshop Implementation of Severe Accident Management Measures, Villigen, Switzerland, 2001	NEA/CSNI/R(2002)12, 2002 NEA/CSNI/R(2001)20, 2001
OECD/CSNI Workshop on Severe Accident Management Operator Training and Instrumentation Capabilities, Lyon, France, 2001	NEA/CSNI/R(2002)11, 2002 NEA/CSNI/R(2001)7, 2001
OECD/CSNI Specialist Meeting on Operator Aids for Severe Accident Management (SAMOA-2), Lyon, France, 1997	NEA/CSNI/R(1997)10, 1997
OECD/CSNI Workshop on Hydrogen mitigation techniques, Winnipeg, Manitoba, Canada, 1996	NEA/CSNI/R(1996)9, 1996
Implementation of hydrogen mitigation techniques during severe accidents in nuclear power plants, a Technical opinion paper	NEA/CSNI/R(1996)27, 1996
OECD/CSNI Specialist Meeting on Severe Accident Management Implementation, Niantic, Connecticut, USA, 1995	NEA/CSNI/R(1995)16, 1995
OECD/CSNI Specialist Meeting on Selected Containment Severe Accident Management Strategies, Stockholm, Sweden, 1995	NEA/CSNI/R(1995)3, 1995
OECD/CSNI Specialist Meeting on Operator Aids for Severe Accident Management and training, Samoa, Halden, Norway, 1993	NEA/CSNI/R(1994)13/A, 1994
OECD/CSNI Specialist Meeting on Instrumentation to Manage Severe Accidents, Cologne, Germany, 1992	NEA/CSNI/R(1993)3, 1993
Specialist Meeting on Instrumentation to Manage Severe Accidents, Cologne, Germany, 1992.	NEA/CSNI/R(1993)3, 1993
OECD/CSNI Specialist meeting on Severe Accident Management Programme Development, Rome, Italy, 1991	NEA/CSNI/R(1992)16, 1992

5. Summary and Conclusions

NEA-CSNI's Working Group on the Analysis and Management of Accidents (WGAMA), through the coordinated undertakings in many significant fields of the severe accidents, has developed a vast pool of information which is readily available to the end-users in the OECD countries. The analysis of the information will produce a valuable track of historical developments in many fields; the phenomenological understanding to the improvements in the treatment of the phenomena in the computer programs; the strengths and deficiencies of the computer programs as they are benchmarked or assessed; implementation of the gained knowledge and experience in practical applications such as implementation in the severe accident management procedures as well as their assessments or even hardware back fittings.

The good coordination by the WGAMA management and excellent support by the member countries providing resources enabling WGAMA to undertake many activities during at least the three decades have produced many products in various forms that have helped the member countries to use the data and to implement the information and to improve their level of experience. The diversity and amount of information that many organizations in the OECD countries receive may not possibly be generated in their countries. Therefore, sharing and co-producing the information has been the vital element for the continued success of WGAMA.