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Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors

IAEA

International Atomic Energy Agency

STATUS AND EVALUATION OF SEVERE ACCIDENT SIMULATION CODES FOR WATER COOLED REACTORS

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INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

After the accident at the Fukushima Daiichi nuclear power plant, the IAEA, in cooperation with the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development, held the International Experts Meeting (IEM) on Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant in Vienna from 16 to 20 February 2015. The objective was to facilitate the exchange of information on R&D activities and to further strengthen international collaboration among Member States and international organizations. One of the meeting's main conclusions was that the Fukushima Daiichi accident had not identified completely new phenomena to be addressed, but had highlighted some areas where the knowledge and understanding of issues associated with severe accidents and other related topics needed to be strengthened. As a follow-up to the IEM, the IAEA organized a meeting on post-Fukushima research and development strategies and priorities from 15 to 18 December 2015. The objective was to provide a platform for experts from Member States and international organizations to exchange perspectives and information on strategies and priorities for R&D regarding the Fukushima Daiichi accident and severe accidents in general. The experts agreed that, to better understand the progression of the Fukushima Daiichi accident, high priority must be given to advancing the current understanding of severe accident phenomenology and to developing, improving and benchmarking severe accident analysis codes.

To address this need, the IAEA organized the Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, held in Vienna from 9 to 12 October 2017, to enable code developers and end users to share experiences and demonstrate state of the art practices. During the meeting, it was observed that severe accident analysts often did not know how to assess the accuracy of their analyses or lacked confidence in the results of their calculations and did not have good appreciation of the sources of uncertainty or variability in their analyses and therefore could not quantify the uncertainties in their predicted results. A total of 37 participants from 19 Member States, together with several IAEA experts, presented state of the art simulation codes addressing severe accidents in water cooled reactors and discussed the need for their improvement, identified gaps and supported IAEA initiatives aimed at launching a new coordinated research project on severe accidents codes analysis. The present publication includes an overview of severe accident codes and their applications, summaries of the presentations and the follow-up discussions, and the meeting's conclusions.

The IAEA acknowledges the contributions of the experts who participated in the Technical Meeting and submitted full papers for presentations, the meeting session chairs and the consultants who drafted and reviewed this publication. The IAEA officer responsible for this publication was T. Jevremovic of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The major nuclear power plant (NPP) accidents at Three Mile Island (TMI) (1979, United States of America), Chernobyl (1986, Union of Soviet Socialist Republics), and Fukushima Daiichi NPP (2011, Japan) have resulted in increased attention on the evaluation of severe accident computational codes and modelling methods, in their improving the models and scopes of simulations, their validation and related uncertainty and sensitivity analysis. A severe accident is defined by the IAEA [1] as an, "Accident more severe than a design basis accident and involving significant core degradation".

The degraded core accident at TMI Unit 2 reached conditions more severe than a design basis accident and prompted new initiatives and subsequent re-evaluation of regulatory processes. The United States Nuclear Regulatory Commission (USNRC) initiated, on 2 October 1980, a "long-term rulemaking to consider to what extent, if any, nuclear power plants should be designed to deal effectively with degraded core and core melt accidents" (USNRC, October 2, 1980). The Fukushima Daiichi NPP accident stressed the necessity to extend the focus of international R&D efforts also to containment phenomena impacting the source term to the environment (including aerosol and core melt behaviour in the containment, risk of combustible gas mixtures).

One of the first attempts to model the phenomena and the way that these phenomena interact in the entire system was the Source Term Code Package (STCP). This code was in fact a loosely and explicitly coupled set of individual codes that modelled separate regimes of a severe accident. Feedback between phenomena was largely not treated in any implicit sense. In response to this early attempt at modelling severe accident progression, the MELCOR code project and the MAAP code development were initiated in the United States in the early 1980s by the USNRC and the Industry Degraded Core Rulemaking (IDCOR) programme, respectively. These codes were among the first fully integrated codes applied to severe accident analysis. The codes represented a significant advance over the STCP in that phenomena occurring within the plant accident progression were coupled to account for the decrease in fuel decay as fission products are released from the fuel for example. Following the development of MELCOR and MAAP, other countries initiated similar integrated code development projects including the ASTEC, ATHLET-CD and SOCRAT in Europe, and more recently the SAMPSON code in Japan. These severe accident computer codes are used to model a range of severe accident phenomena such as thermal hydraulics, heating, hydrogen generation and combustion, reactor vessel failure, core melting, molten core–concrete interactions, containment performance and fission products release (more detailed information is included within Section 2).

In the decades following the TMI Unit 2 accident, the codes were used largely in what is commonly termed a 'deterministic mode' where single representative accidents were modelled to represent classes of accident such as unrecovered large and small break loss of coolant accidents (LOCAs) or station blackout (SBO). During this time, the analyses performed with these tools were computationally intensive calculations carried out on much slower computers with much lower memory in comparison to modern computational platforms. Uncertainty in the operative physics/phenomena and the stochastic aspects of accident conditions in these types of analyses was known to exist but onerous to quantify. For this reason, the deterministic analyses were often biased conservatively in hopes of producing a bounding calculational result which could be compared to the requirements such as for example public exposure limits. The NUREG-1150 study included in the probabilistic risk assessment methodology some estimation of uncertainty in severe accident progression but made heavy use of expert elicitation to estimate uncertainty in key figures of merit such as percent of core metal oxidized. Code analyses were largely impractical at the time of the NUREG-1150 project. Code stability and execution failures were also significant impediments to producing large numbers of analyses that might express the variability in predicted outcomes.

In ensuing years as the severe accident codes improved in robustness and runtime efficiency and as computational platforms significantly increased in speed, sampling based uncertainty studies began to emerge using sampling methodologies embodied in statistical tools such as DAKOTA, SUSA, SUNSET and MELCOR–Uncertainty Engine. These tools investigated the uncertainties in an analysis to be expressed in terms of variability in the code input and boundary conditions that could be 'propagated' through the severe accident analysis producing an ensemble of 'answers' from which probability distributions instead of single realization point values. In this way a likelihood distribution of accident figures of merit is obtained that give indications of mean values, central tendencies and dispersion in the answers. Early analyses were heroic owing to the computational challenges and machine limitations. Especially after the 2011 Fukushima Daiichi NPP accident, nuclear experts have intensified the evaluation of severe accidents with increased attention to severe accident computational codes and modelling methods.

Nuclear power plant safety systems are designed to mitigate a range of atypical operating conditions. Defined as "accident conditions more severe than a design basis accident" and "involving significant core degradation", severe accidents are beyond design accidents — low probability but high impact. In these highly unlikely events, computer codes are used to model a wide range of associated phenomena, thermal hydraulics, heating, hydrogen generation and combustion, reactor vessel failure, core melting, molten core–concrete interaction, containment performance, and fission product release. These codes are robust and computational platforms assure execution on massively parallel computational resources with thousands of individually addressable processors; the sampling based uncertainty methods are now easily within reach of severe accident analysis efforts. Examples of this include recent uncertainty analysis studies performed by the USNRC and Sandia National Laboratories in the State of Art Reactor Consequence Analyses (SOARCA). Likewise, other uncertainty and sensitivity analysis demonstrations have been accomplished by ASTEC and RELAP/SCDAPSIM on evaluation of the QUENCH, QUENCH–6 and QUENCH–3 experiments.

As an outcome of the IEM on Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant and the Technical Meeting on Post-Fukushima Research and Development Strategies and Priorities, R&D efforts to further understand severe accident phenomenology and to develop/improve benchmarks for severe accident analysis codes were prioritized.

The Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors was held in Vienna, 9–12 October 2017, in order to provide a platform for detailed presentations and technical discussions on the status and evaluation of severe accident simulation codes for WCRs. Furthermore, the meeting provided a forum for Member States to exchange knowledge on code innovations, limits and gaps to gather information for collaboration on these aspects. The meeting also included a special session on modelling of the Fukushima accident. Specific objectives of the meeting were to:

- Review and discuss the status and the recent progress in deterministic simulation codes for evaluation of severe accidents in WCRs;
- Collect information on the needs for code improvements, validation and verification, in addition to uncertainty assessments in assessing the needs for new benchmark models (in reviewing types of experimental and numerical benchmarks);
- Update on the most recent simulation models and results on the Fukushima accident;
- Assess information on the available PC based basic principle simulators that cover severe accidents in WCRs and identify the needs for their further development;
- Discuss the prospects of future R&D and projects on advancing the deterministic simulation codes for severe accident evaluation in current and advanced WCRs.

The meeting programme included presentation and discussion sessions to enable participants to contribute to the summary and highlights of the meeting, and to make recommendations to the IAEA on the future activities in severe accident simulation codes in WCRs. The technical papers and discussions supported the objective to complete a comprehensive review of the status and progress in severe accident simulation codes, benchmark models, and how these codes can be improved with future research and development. The meeting focused on the most commonly used codes for severe accident analysis with participants highlighting code features and limitations, with the key observations regarding the state-of-the-art practices in severe accident analysis, including uncertainty and sensitivity analyses. The appropriate use of plant simulators for severe accident analysis was also discussed. An overview of the latest developments in the comprehensive modelling of the accident at Fukushima Daiichi NPP was presented and helped to provide important considerations in severe accident modelling and recommendations documented in this TECDOC. The meeting participants included end users and developers in the area of severe accident modelling. The full papers as provided by the participants and their respective presentations during the meeting are included in the CD-ROM.

1.2. OBJECTIVES

The objective of this TECDOC is to capture the current state-of-the-art knowledge on severe accidents codes status by summarizing the information from the Technical Meeting on Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors that was held in Vienna, Austria on 9–12 October 2017. The information is detailed in terms of major findings, identified gaps and recommended future actions with the purpose to capture the current state-of-the-art related to severe accident simulation codes for WCRs.

1.3. SCOPE

This TECDOC is intended to represent an objective summary of reference information for interested organizations, individuals, and decision makers from countries embarking on, or considering implementation of, new nuclear power programmes as well as from those expanding their existing programmes.

1.4. STRUCTURE

Section 1 of this publication describes the background, objectives and its structure. Section 2 provides a detailed summary of the description of the severe accident codes and models as well as provides developer submitted code descriptions and questionnaire responses. Section 3 provides an overview of the Technical Meeting sessions on the topics of (1) computer codes and models for evaluation of severe accidents in water cooled reactors, (2) new developments in comprehensive computer modelling of the Fukushima Daiichi NPP accident and (3) basic principle simulators for severe accidents. In Section 4, the general conclusions and recommendations taken from the Technical Meeting are summarized.

Annex I provides a detailed summary of presentations and discussion from the Technical Meeting. Annex II details the contents of the accompanying CD-ROM, included as part of this TECDOC with papers and presentations provided at the meeting sessions. Annex III lists and briefly describes IAEA publications related to severe accident analysis. Annex IV lists and describes some of the IAEA projects and activities related to severe accident codes and analyses.

2. DESCRIPTION OF SEVERE ACCIDENT CODES AND MODELS

This section provides descriptions of severe accidents codes and their various applications. Section 2.1 discusses an overview of the physics and phenomena operative in severe accidents within the NPP including a description of the key systems and general chronology of severe accident events modelled in the system level computer codes. Section 2.2 provides a description of commonly used severe accident codes and practical information on obtaining and using these codes. Section 2.3 provides some key observations and insights from the various applications that were presented in the papers during the Technical Meeting on Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors held in October 2017 in Vienna, Austria.

There are four fundamentally different types of applications of severe accident computer codes:

— *Deterministic application:* refers to single scenario accident sequence analyses and is probably the most common application of severe accident codes. These analyses are generally a user's best estimate model for a representative accident sequence in a NPP, such as an unrecovered large break loss of coolant or station blackout analysis. Such analyses provide insight into general accident progression signatures for various figures of merit such as containment performance, hydrogen generation or fission product releases.
— *Probabilistic applications:* generally involve conducting a larger number of analyses to reflect a spectrum of different potential severe accidents as might be performed in a risk study such as a Probabilistic Safety Analysis (PSA), or in an uncertainty analysis where permutations of a given accident sequence are examined using sampling methods such as Monte Carlo to capture variability in predicted accident outcome.
— *Forensics application:* In the application of severe accident codes to accident reconstruction (for example, in reconstructing the Fukushima accident), codes are used in a 'forensics' mode where known or suspected accident events such as core slumping, vessel failure or containment failure events are imposed on the code in an attempt to synchronize observations with code predicted behaviour and better identify plausible accident progression trends.
— *Prognostic/decision support application*: Severe accident codes are finding application in training and decision support roles for example in assessing transition from emergency operating procedures to severe accident management guidelines and in forecasting potential radiological releases from postulated accidents under anticipated weather trends in order to inform potential public protective actions such as sheltering or evacuation.

2.1. GENERAL DESCRIPTION OF SEVERE ACCIDENT PHENOMENA IN SEVERE ACCIDENT ANALYSIS

Severe accidents in NPPs encompass a very wide range of interacting phenomena:

— Thermal hydraulics behaviour in the reactor core and coolant loops;

— Degradation of the reactor core, including oxidation of fuel rod cladding, melt formation, relocation of material to the lower part of the core, melt pool behaviour, in-vessel failure, ex-vessel corium recovery, molten core–concrete interactions;
— Release of fission products from the fuel, structural material release, transport and deposition in the primary circuit, their behaviour in the containment (with special emphasis on iodine and ruthenium), aerosol behaviour;
— Thermal hydraulics in the containment, hydrogen behaviour, molten fuel–coolant interactions, direct containment heating.

Depending on the core initial state and the accident scenario, core uncovery could be reached and leads to heat up of uncovered fuel due to residual decay heat; clad (fuel sheath) deformation and failure; oxidation of metals (e.g. Zr) by steam and exothermic reaction which accelerates the core degradation and release of large quantities of flammable H2/D2 into the containment; and chemical interactions amongst all the materials, leading to core melting and liquefaction.

2.1.1. In-vessel phenomena

After the start of the transient (SOT), the initial phase of the accident progression is dominated by thermal hydraulics phenomena and processes. After this phase, dependent on the postulated transient such as large break LOCA (LBLOCA), small break LOCA (SBLOCA) or SBO, and reactor type (pressurized water reactor (PWR), VVER, boiling water reactor (BWR), RBMK, pressurized heavy water reactor (PHWR)), the severe accident progression could evolve considering the different boundary conditions and mitigation actions that are supposed to fail during the accident evolution. It is to underline that the time windows between the SOT and the core boiloff and consequent uncovering depend on the postulated accident scenario. Therefore, in general, a severe accident code should have the capabilities to accurately predict the thermal hydraulics transient progression and the related phenomena and processes (e.g. single and two phase natural circulation, heat transfer in a covered/partially uncovered core, heat transfer in steam generator primary and secondary side, break flow) in current and advanced reactor designs. The subsequent core degradation phases are qualitatively dominated by a common series of phenomena/processes, as briefly described in this section.

In general, the decrease in the reactor core water level leading to the uncovering of the top of active fuel, is due to a steady evaporation of the reactor coolant for the boiloff phenomena taking place during this first phase of a severe accident progression. The heat transfer in uncovered core determines a reduction of the energy removed from the core with a consequent increase of the temperature of the fuel rods. The steam formed in this phase interacts with the different materials present in the core, such as Zircaloy and steel, leading to exothermic oxidation reaction. In this early phase of the severe accident, once temperatures of 1300 K are reached, the Zircaloy exothermic oxidation process starts to be important (steel oxidation is in general less significant). Its significant energy release, added to the decay heat, determine a temperature escalation with a consequent heat up of the core and hydrogen generation. Considering the plants investigated and the transient progression, the inception of the core uncovery could take different time from several minutes to several hours. Depending on the transient progression a high pressure or low pressure scenario could take place in the reactor core.

The subsequent core degradation and melt progression phase determine a loss of the core geometry and, as a consequence, a change of the reactor coolant flow path. This phase starts with the degradation of the core materials with lowest melting temperatures and it is followed by the fuel and cladding degradation and relocation. The hydrogen generated during this phase is dependent from the core degradation progression and the consequent available area for the oxidation and flow blockage phenomena. In this late phase of the severe accident, additional hydrogen could be generated, due to the oxidation phenomena taking place when the degraded core material massively relocates by slumping into the lower part of the reactor pressure vessel (i.e. plenum for PWR, calandria vessel for PHWR). The long term phenomenological behaviour is dominated by physical and chemical phenomena, characterizing the degraded core material in the lower plenum, and the lower head boundary condition (e.g. cavity flooded with water). This determines the time of lower head failure. During the in-vessel phase, fission product release and transport take place due to the core degradation phenomena occurring during the severe accident progression.

Detailed information about all the thermal hydraulic phenomena characterizing transients before degradation of the core can be found in the internationally recognized OECD/NEA/CSNI thermal hydraulics ITF and SETF validation matrix [2,3,4]. In relation to advanced reactor designs, the thermal hydraulic phenomena of relevance are investigated in detail in [5,6,7,8]. In relation to the in-vessel core degradation phenomena, more detailed information about all the core degradation phenomena characterizing these transients can be found in the OECD/NEA/CSNI internationally recognized In-vessel Core Degradation Code Validation Matrix [9]. The current state of knowledge in core melt accident is investigated in [10]. More information about accident postulated source terms for LWR can be found in [11]. Important in-vessel phenomena are also described in [12].

2.1.2. Ex-vessel phenomena

After failure of the reactor vessel, melt is relocated from the reactor vessel to the containment. A low pressure scenario should be distinguished from a high pressure scenario. In the latter the question of direct containment heating (DCH) may be an issue, which is specific to the geometrical flow paths and available melt entrapment mechanisms in the lower compartments of the containment. For several reactor designs, it has been concluded that the risks of DCH for containment integrity are small. For the further progression of the accident it is important to distinguish between an initially dry cavity or a flooded cavity. In the first case under low pressure, corium will spread driven by gravity. Maximum extent of spread corium will depend on the physical state of the corium at time of ejection from the reactor core as well as on the mass flow boundary conditions. In the case of a wet cavity, energetic fuel coolant interactions (FCI) could threaten the containment integrity. Spikes in steam production rate will contribute to the pressure built up.

During melt ejection into deep water pools, there is some potential that the compact melt jet may fragment and form a coolable debris bed submerged in water in the containment. If the degree of fragmentation is not sufficient in case of a wet cavity or in a dry cavity, the relocated melt would finally start to ablate the concrete, which could take several hours up to several days, depending on concrete thickness and composition. During this long term interaction non-condensable gases as well as steam will be released. This may have effect on the risk of deflagration of combustible gas mixtures in the containment. The melt might be flooded as part of an accident management measures. Several mechanisms (bulk cooling, water ingression into corium top crust, melt entrainment and eruption) have been identified to contribute to the cooling of the melt during the top flooding phase of the molten core concrete interaction (MCCI). Hydrogen production during MCCI may be significant in terms of absolute masses; rates are quite comparable to in-vessel production.

The risk of large spatial accumulation of combustible gas mixtures, and thus the risk of fast burns, might be mitigated by presence and operation of igniters and passive autocatalytic recombiners (PARs) positioned effectively in the containment compartments. Released gases and fission products would distribute in the containment according to their physical and chemical forms. Volatile gases will freely distribute due to the atmospheric flow conditions in the containment. Aerosols might be transported with atmospheric flows until they agglomerate and settle. Deposited aerosols might be washed off from structures by condensing steam/water films along the structures and relocated to the sump of the containment. Spray systems may wash out aerosols from the atmosphere but may also re-inject aerosols, if water is taken from a contaminated reservoir. The concentration of gaseous iodine in the atmosphere does heavily depend on chemical reactions: in the sump, with decontamination paint etc.

The source term to the environment depends at last on the release path or failure mode of the containment: containment failure due to overpressure, release via filtered/unfiltered venting paths, release via penetration of corium through the concrete foundation.

A comprehensive description of ex-vessel phenomena relevant to the safety of LWR during severe accidents can be found in [13]. In those references, all the phenomena listed above are addressed. A more detailed description of the behaviour of nuclear aerosols in the containment is given in [14]. The current state of knowledge of the topic molten corium–concrete interaction is outlined in more detail in [15].

2.2. COMMONLY USED SEVERE ACCIDENT CODES

There are several computer codes which could be used (or are dedicated) for the analysis of processes during severe accidents in water cooled reactors. In this section, computer codes which were presented during the Technical Meeting on Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors held in October 2017 in Vienna, , are presented in more detail. The following code descriptions are provided by code developers and include developer responses to a common questionnaire regarding the codes' general information, software licensing information, CPU requirements, status of related documentation and availability of user support.

2.2.1. AC²

AC² is a coupled code system mainly developed by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH for the simulation of design basis and severe accidents in LWR and consists of three main modules:

- The ATHLET module is used for the detailed thermohydraulic simulation of transients and loss of coolant accidents of the whole reactor coolant system.
- The ATHLET-CD module is the extension of ATHLET for the simulation of severe accidents inside the reactor coolant system and the spent fuel pool.
- The COCOSYS module is used for the detailed simulation of accidents and severe accidents in the containment/buildings of LWR.

The main code features of ATHLET are an advanced thermal hydraulic modelling (compressible fluids, mechanical and thermal nonequilibrium of vapor and liquid phase) and heat generation, heat conduction and heat transfer to single or two phase fluid considering structures of different geometry, e.g. rod or pebble bed. Diverse fluids can be simulated: light or heavy water, helium, sodium, lead or lead-bismuth eutectic. Interfaces exist to specialized numerical models such as 3D neutron kinetic codes or 3D CFD codes for coupled multi-physical or multiscale simulations.

ATHLET has been developed and validated to be applied for all types of design basis and beyond design basis accidents without core damage in LWR, like PWR, BWR, VVER, and RBMK. The validation is mainly based on pre- and post-test calculations of separate effects tests, integral system tests including the major International Standard Problems, as well as on real plant transients.

The main code features of ATHLET-CD are related to the reactor coolant system response during severe accidents. This includes core damage progression, fission product and aerosol behaviour, source term calculation for containment analyses, and evaluation of accident management measures.

The rod module in ATHLET-CD consists of models for fuel rods, absorber rods (AIC and B4C) and for the fuel assemblies including BWR canisters and absorbers. The module describes the mechanical rod behaviour (ballooning), zirconium and boron carbide oxidation, Zr–UO2 dissolution as well as melting of metallic and ceramic components. The model allows oxidation, freezing, re-melting, re-freezing and melt accumulation due to blockage formation. The feedback to the thermal-hydraulics considers steam starvation and blockage formation in the core. Besides the convective heat transfer, also the energy exchange by radiation between fuel rods and to surrounding core structures is considered by ATHLET-CD. The nuclide inventories are calculated by a pre-processor (OREST) as a function of power history, fuel enrichment and initial reactor conditions. The release and the transport of nuclides consider decay heat (α, β, γ) and further decay by means of mother–daughter chains. For the simulation of debris beds a specific model is under development. The transition of the simulation of the core zones from the rod module to the debris bed model depends on the degree of degradation in the zone. The code development comprises also late phase models for core slumping, melt pool behaviour within the vessel lower head as well as for vessel failure.

The validation of ATHLET-CD is based on integral tests and separate effect tests, proposed by the CSNI validation matrices, and covers thermal hydraulics, bundle degradation as well as release and transport of fission products and aerosols.

The main code features of COCOSYS are related to the containment/building response during accidents and severe accidents. Essential phenomena and interactions between the individual processes, like between thermal hydraulics and hydrogen combustion as well as fission product and aerosol behaviour, are treated in a comprehensive way. COCOSYS provides also the separate LAVA code for the simulation of the core melt spreading and relocation in the containment after reactor pressure vessel failure.

The COCOSYS thermal hydraulic module covers different zone and junction models required to describe the physical state of the containment atmosphere adequately, including also the presence of water pools. For an adequate simulation of the different systems or boundary conditions, specific models are implemented, like rupture discs, atmospheric valves, flaps/doors and specific pressure relief valves used in Russian types reactors. For the simulation of water drainage, several models are available, describing the sump balance, water flow through pipes and along walls. The implemented pump system model is flexible enough to simulate complete cooling systems, e. g. emergency core cooling systems.

Structure objects consider heat transfer to walls, floors and ceilings of the building. Structure objects can be partly submerged by water. The heat exchange between structures and their assigned zones are calculated via convection, condensation or radiation (including wall to wall) heat transfer correlations. It is possible to simulate different types of coolers (incl. atmosphere cooling systems), spray systems, ventilation systems, ice condensers and PARs. A simplified model is used to simulate hydrogen combustion and flame propagation between different compartments without requiring much additional user input. The calculated combustion rates of hydrogen and deflagration velocities in the respective zones depend on several empirical correlations which include empirical parameters.

The nuclide behaviour model considers the reactor's initial core inventory and calculates on this basis the decay of the fission products according to the time of the onset of the release by using established nuclide libraries (analogous to ORIGIN).

The aerosol model distinguishes between soluble and insoluble as well as hygroscopic and non-hygroscopic aerosols. The following deposition processes are covered: sedimentation, diffusive deposition, thermophoresis and diffusiophoresis.

Iodine chemistry considers approximately 70 different reactions. It distinguishes between 17 iodine species in the atmosphere and 14 iodine species in the sump. It calculates iodine transport between atmosphere and sump as well as across the compartments.

All relevant processes relating to the fission products and the different carriers are considered: Deposition of aerosol particles on surfaces and in the sump by natural processes or aided by technical systems such as filters and spray systems, washdown from walls and washout by spray, and carrier change due to radioactive decay. There are special models for the simulation of filters (HEPA filters, granulate filters) implemented in COCOSYS. The retention of aerosols during gas transport through water pools is calculated by the SPARC-B module. This allows among other things the simulation of 'pool scrubbing' in the pressure suppression system of a boiling water reactor.

Molten corium concrete interaction (MCCI) is calculated based on a lumped parameter approach with layer averaged heat and mass balances. Generally, multiple melt pools with interactions between molten corium and containment structures (sidewall, floor) of up to two melt layers (oxide and metal) in each pool may be defined. The 2D cavity geometry is in principle axisymmetric and is determined as a function of time using the local energy conservation at each boundary node and a common melting approach (Stefan's equation). Fission product release from the MCCI pool is approximated assuming thermos dynamical equilibrium between gas bubbles released from concrete decomposition and the melt.

COCOSYS is validated on a wide spectrum of separate and integral experiments performed at German or international test facilities. The experiments performed in the former Battelle Model Containment (BMC) and the former Heiß Dampf Reaktor (HDR) as well as the ongoing tests in the THAI facility represent a strong pillar of the COCOSYS validation.

AC^2 includes the German nuclear plant analyser ATLAS. The ATLAS environment allows not only a graphical visualization of the calculated results but also an interactive control of simulation.

The development and the validation of AC^2 are funded by the German Federal Ministry of Economics and Technology (BMWi) on the basis of several resolutions of the German Bundestag.

A summary of the code specifics is given in Table 1.

TABLE 1. SPECIFICS OF AC^2 CODE

GENERAL INFORMATION	
Code name including acronym (and current version)	AC^2 (ATHLET 3.1.A patch 4, ATHLET-CD 3.1.A patch 4, COCOSYS 2.4 V5)
Developing organization	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH
Severe accident applicability (e.g. BWR, PWR, PHWR, spent fuel pool, dry storage, etc.)	PWR, BWR, spent fuel pool.
Describe the capability for user defined functions transients available? (e.g. Control functions, user defined logic functions, etc.)	Arbitrary control functions can be defined by the user by a general control module, which is part of AC^2. Several user defined models can be provided as plugins.
What is the nodalization approach of the code? (e.g. free nodalization, pre-defined nodalization)	A lumped parameter approach is used, which offers a widely free nodalization.
Capability to interface with special codes? (e.g. gas flow, CFD, atmospheric dispersion, Origen, etc.)	Dedicated coupling interfaces to ANSYS CFX, OpenFOAM and the containment water pool model CoPool. General coupling interface, which can be adapted by the user to couple to arbitrary other codes (e.g. structural mechanics codes). Interfaces to the neutron flux codes Quabox/Cubbox, KIKO3D and BIPR.
Which code features allow users to conduct/explore uncertainty and sensitivity analysis? If yes, is this functionality integrated or separate (e.g. integrated, external tools such as Dakota, SUSA, SUNSET, RAVEN, etc.)	Uncertainty and sensitivity analysis can be performed with the separate GRS tool SUSA.
SOFTWARE LICENSING INFORMATION	
Software licensing organization	GRS
Who are the intended users? (if any: regulators, industry, etc.)	TSOs, regulators, research institutes. Under some constraints also industry.
Who can receive a license to your code? (organization/cluster/individual/individual computer)	All above-mentioned Organizations can apply for a license.
Is the source code available? What is the agreement for receiving the code?	No, except for partners of code development collaborations. Positive evaluation of the request by (1) technical assessment and (2) by German Federal Bureau for Economic Affairs and Export Control. No further transfer of the code to third party; periodic feedback to GRS (yearly).
Please provide a general process for prospective users to apply for the code:	Official request to the head of Safety Research Division of GRS (see https://user-codes.grs.de/code-transfer). Basic requirements to get a AC^2 license is the permission of the German export control BAFA and the signing of a Software License Agreement and a Code Certificate.
CPU REQUIREMENTS	
What CPU platform does your code work on (Windows, Mac, Linux)?	Windows, Linux
Compiler	AC^2 is delivered as executables, so that no compiler is needed (Intel Visual Fortran & C++)
High performance computing capabilities	AC^2 runs on Linux HPC computer. The module ATHLET-CD is parallelized based on OpenMP. The parallelization of the COCOSYS module is planned in future.

TABLE 1. SPECIFICS OF AC^2 CODE (cont.)

STATUS OF DOCUMENTATION	
Is there any documentation on installation instructions?	Installation Manual; installation on Windows platform is performed via install shield.
Describe available documentation available to users (e.g. user guides, input requirements, modelling guidance, validation)	User manual (including section on guidelines); reference manual; assessment manual (including state of validation); regression test report. Documentation in English is delivered with the code to licensee and is separate for each of the 3 modules.
Is documentation publicly available?	The AC^2 documentation is largely not published. However, project reports on code development and validation (mostly in German) are available.
SUPPORT	
What additional resources are available for users? (e.g. technical support, bug reporting, etc.)	Documentation and additional information as bug report and user hints are available on the user-area: https://user-codes.grs.de. GRS offers preferential technical support contracts.
Is there a user group for the code?	Periodic user meetings are organized by GRS.
Describe any training workshops available for users from the code developer:	— Participation to training is recommended to licensees (such workshop would be subject of separate and individual contracts). — General introduction into a AC^2-module (COCOSYS, ATHLET or ATHLET-CD basic training): 1 week. — Intensive training for DBA analyses: 7 trainings each with 3 weeks including homework further trainings on licensee’s request.
Are test cases or examples available for user validation of their installation?	Yes, installation test cases are available.
Are reference plant models or input decks available to users?	In the user manual several examples with input for safety systems are provided. The intensive training course comprises the setup of a specific input deck based on licensee’s facility data.
Briefly describe what future, near term developments will be implemented into the code:	ATHLET-CD: Implementation of an asymmetric core nodalization; improvement of the lower plenum models. COCOSYS: Redesign of Aerosol and FP module to harmonize all FP/aerosol and iodine models; homogenization of handling of geometrical structures in thermal hydraulic and FP/aerosol models. AC^2: Consolidation of code coupling between main modules.
Describe any supporting software for I/O processing or a GUI:	— GUI to control the calculations. — ATLAS, an interactive GUI interface which can be applied ‘on-line’ for simulations with interaction on the course of the accident sequence or ‘off-line’ as post processor. — Further different I/O-software to support the user.

TABLE 1. SPECIFICS OF AC^2 CODE (cont.)

SUPPORT	
Representative Publications	— ALLELEIN, H.-J., ARNDT, S., KLEIN-HEẞLING, W., SCHWARZ, S., SPENGLER, C., WEBER, G., COCOSYS: Status of development and validation of the German containment code system, Nucl. Eng, Des. **238** 4 (2008) 872, 889. — REINKE, N., KLEIN-HEẞLING, W., SPENGLER, C., SCHWARZ, S., BECK, S., NOWACK, H., SONNENKALB, M., "Development, validation, and application of the containment code system COCOSYS", 11th Intl. Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Operation and Safety (NUTHOS-11), Gyeongju, 2016. — BUCHHOLZ, S., KLEIN-HEẞLING, W., BONFIGLI, G., KACZMARKIEWICZ, N., NEUKAM, N., SCHÄFER, F., WAGNER, T., "The code system AC^2 for the simulation of advanced reactors within the frame of the German EASY project", 17th Intl. Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-17), Xi'an, 2017. — LOVASZ, L., WEBER, S., SCHÖFFEL, P., PANDAZIS, P., AUSTREGESILO, H., "Status of Development of GRS Code System AC^2, Part I: Modelling of Reactor Phenomena", IAEA Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, Vienna, 2017. — SPENGLER, C., ESCHRICHT, D., KLEIN-HESSLING, W., ARNDT, S., BAKALOV, I., NOWACK, H., BECK, S., REINKE, N., SONNENKALB, M., "Status of Development of GRS Code System AC^2, Part II: Modelling of Containment Phenomenon", IAEA Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, Vienna, 2017.

2.2.2. ASTEC code

The ASTEC (Accident Source Term Evaluation Code) software system makes it possible to simulate all phenomena that take place during a water cooled reactor meltdown accident, from the initiating event to the discharge of radioactive materials (called the source term) out from the containment. This particularly covers western designed pressurized water reactors such as those used for France's nuclear power supply, Russian designed PWRs (VVERs), boiling water reactors and pressurized heavy water reactors. ASTEC is currently maintained and developed by the IRSN.

The main applications of the software package are safety analyses for nuclear reactors (e.g. the European Pressurized Reactor, EPR), source term evaluations in accident situations and the development of severe accident management guidelines. ASTEC is widely used in IRSN level 2 probabilistic safety assessments for French NPPs. It is also used to prepare and interpret the experimental programs with respect to either in-vessel or ex-vessel phenomena which could occur during meltdown accidents. Furthermore, ASTEC is also occasionally used for preparing certain emergency exercises (nuclear or radiological accident simulation exercises for training various players and the organization itself) at the IRSN Crisis Technical Centre.

General structure: ASTEC covers the entire phenomenology of severe accidents except steam explosion and the mechanical integrity of the containment. Its modular structure simplifies qualification by comparing the simulated results with those obtained experimentally.

The general structure of the code is illustrated in Figure 1 (page 16) and its specifics are provided in Table 2.

Main models: Each module of the code handles phenomena that occur in a part of the reactor or phase of the accident, and in particular:

- The two phase thermal hydraulics of coolant flows in the reactor coolant primary and secondary systems including a 2D description of the vessel;
- The degradation of materials within the vessel, when the temperatures reached under the effect of the core's residual power exceed a threshold leading to significant oxidization of the fuel rod claddings due to water vapor as well as various chemical interactions between the materials that make up either the fuel rods or the control rods. This may go as far as the materials melting, leading to the formation of a mixture of melted materials known as corium. Another possibility is the formation of solid debris from the premature fragmentation of fuel pellets combined with the embrittlement of fuel rod claddings. The behaviour of corium, once located in the vessel bottom after its slumping from the core, is modelled accounting dynamically for the possible stratification of materials to form several metal and oxide layers. This behaviour is modelled until the vessel bottom barrier is either breached or the corium layers achieve to be stabilized on the inside if the vessel can be cooled from the outside (Figure 1);
- The release of fission products (FP), particularly iodine, from fuel in the core;
- The transport of FPs and aerosols as well as their physical and chemical behaviour in the primary and secondary cooling systems, then in the containment. Special attention is particularly paid to the behaviour of the many iodized species in its various forms (molecular iodine, gaseous organic iodides, iodine oxides aerosols, etc.);
- The thermal-hydraulics within the containment using a 0D volumes approach, classically called ‘lumped parameter code’;
- The erosion of the vessel shaft's raft by corium located there in the event that the vessel is breached, taking into account possible subsequent arrival of further. This basemat erosion, called corium–concrete interaction (CCI) is modelled using a volume based approach or single-dimensional layers. The model makes it possible to process the conditions of a dry CCI or a CCI under water, with the second situation including the cooling of corium by submersion in water based on the procedures for managing severe accidents;
- DCH by the transfer of hot gases and corium droplets from the reactor cavity, following the rupture of the vessel;
- Combustion of hydrogen or carbon monoxide accumulated within the containment and the associated risk of explosion.

Furthermore, ASTEC evaluates the radioactivity of isotopes and the associated residual power in all parts of the reactor, as well as dose rates in the containment.

FIG. 1. General structure of the ASTEC code (reproduced courtesy of Institut de Radioprotection et de Sûreté Nucléaire — IRSN, all rights reserved).

TABLE 2. SPECIFICS OF ASTEC CODE

GENERAL INFORMATION	
Code name including acronym (and current version)	ASTEC V2.1
Developing organization	IRSN
Severe accident applicability (e.g. BWR, PWR, PHWR, spent fuel pool, dry storage, etc.)	PWR, BWR, PHWR, spent fuel pool.
Describe the capability for user defined functions transients available? (e.g. Control functions, user defined logic functions, etc.)	Multiple built in control functions (arithmetic, logical, user-defined) as well as vectorized control functions.
What is the nodalization approach of the code? (e.g. free nodalization, pre-defined nodalization)	Free nodalization.
Capability to interface with special codes? (e.g. gas flow, CFD, atmospheric dispersion, Origen, etc.)	Atmospheric dispersion codes.
Which code features allow users to conduct/explore uncertainty and sensitivity analysis? If yes, is this functionality integrated or separate (e.g. integrated, external tools such as Dakota, SUSA, SUNSET, RAVEN, etc.)	PROMETHEE, R, SUNSET

TABLE 2. SPECIFICS OF ASTEC CODE (cont.)

SOFTWARE LICENSING INFORMATION	
Software licensing organization	IRSN
Who are the intended users? (if any: regulators, industry, etc.)	Regulators, industries, TSO, researchers.
Who can receive a license to your code? (organization/cluster/individual/individual computer)	Organization, individual computer.
Is the source code available?	No
What is the agreement for receiving the code?	License agreement and export control procedures.
Please provide a general process for prospective users to apply for the code:	Contact the Severe Accident Department, project owner of ASTEC: IRSN/PSN-RES/SAG
CPU REQUIREMENTS	
What CPU platform does your code work on (Windows, Mac, Linux)?	Windows, Linux
Compiler	Intel 2017
High performance computing capabilities	OpenMP (under conditions)
STATUS OF DOCUMENTATION	
Is there any documentation on installation instructions?	Yes, delivered with each version.
Describe available documentation available to users (e.g. user guides, input requirements, modelling guidance, validation)	Users guide, online user manual, physical and numerical documentation, validation documentation.
Is documentation publicly available?	—
SUPPORT	
What additional resources are available for users? (e.g. technical support, bug reporting, etc.)	User support, bug reporting, version delivering on: https://gforge.irsn.fr/gf/project/astec/
Is there a user group for the code?	Yes (ASCOM Nugenia TA 2 project)
Describe any training workshops available for users from the code developer:	Standard beginner training (5 days, all modules and phenomena addressed). Specialized training: on demand.
Are test cases or examples available for user validation of their installation?	Yes (delivered with versions).
Are reference plant models or input decks available to users?	Yes (reference plants distributed for PWR 3 loops, KONVOI, BWR, VVER, CANDU and SFP).
Briefly describe what future, near term developments will be implemented into the code:	Extension of capabilities as concern EPR, BWR, VVER design. Ease-of-use improvements.
Describe any supporting software for I/O processing or a GUI:	Embedded ODESSA graphical tools; Graphical editor (under development).
Representative Publications	— LABORDE, L., CARENINI, L., FICHOT, F., “External Reactor Vessel Cooling modeling in ASTEC V2.1 code”, NUTHOS-18. — CHATELARD, P., BELON, S., BOSLAND, L., CARÉNINI, L., COINDREAU, O., COUSIN, F., MARCHETTO, C., NOWACK, H., PIAR, L., CHAILAN, L., Main modelling features of ASTEC V2.1 major version, Ann. of Nucl. Energy **93** (2016) 83, 93. — BONNEVILLE, H., CARÉNINI, L., BARRACHIN, M., Core melt composition at Fukushima Daiichi: results of transient simulations with ASTEC, Nucl. Tech. **196** 3 (2016) 489, 498. — NOWACK, H., CHATELARD, P., CHAILAN, L., HERMSMEYER, S., SANCHEZ, V.H., HERRANZ, L.E., CESAM – code for European severe accident management, EURATOM project on ASTEC improvement, Ann. Nucl. Energy **116** (2018) 128, 136. — CARÉNINI, L., FICHOT, F. SEIGNOUR, N., Modeling issues related to molten pool behavior in case of in-vessel retention strategy, Ann. Nucl. Energy **118** (2018) 363, 374.

2.2.3. MAAP5 code

MAAP5 is a Modular Accident Analysis Program, Version 5, a computer code that simulates the response of LWR and CANDU power plants during severe accidents. MAAP5 treats the full spectrum of important phenomena that could occur during an accident, simultaneously modelling those that relate to the thermal hydraulics and the fission products. It also simultaneously models the primary system, core, containment, and reactor/auxiliary building. Thus, given a set of initiating events and operator actions, MAAP5 predicts both the thermal hydraulics and fission product response of the entire plant as the accident progresses. For these reasons, MAAP5 is often referred to as an integral severe accident analysis code.

The purpose of MAAP5 code is to provide an accident analysis that can be used with confidence by the nuclear industry in all phases of severe accident studies, including accident management, for current reactor/containment designs and for advanced LWRs and that can be used to do the following:

- Predict the timing of key events (for example, core uncovery, core damage, core relocation to the lower plenum, and vessel failure);
- Evaluate the influence of mitigative systems, including the impact of the timing of their operation;
- Evaluate the effects of operator actions;
- Predict the magnitude and timing of fission product releases;
- Investigate uncertainties in severe accident phenomena;
- Investigate spent fuel pool (SFP) accident scenarios;
- Calculate in-plant and ex-plant radiation doses using MAAP5-DOSE.

MAAP5 results are primarily used to determine Level 1 and 2 success criteria and accident timing for probabilistic risk assessment analyse (PRAs). They are also used for investigating accident management strategies, equipment qualification analyses, fission product large early release frequency (LERF) determinations, integrated leak rate test evaluations, emergency planning and training, simulator verification, analyses to support plant modifications, generic plant issue assessments (such as significance determinations), and other similar applications.

Parallel versions of MAAP5 support BWRs and PWRs. Other unique versions of the MAAP5 code exist for CANDU, VVER, and advanced thermal reactor designs. In addition, MAAP5 is applicable to both current and advanced LWR designs, with models that represent the passive features of the latter.

The MAAP5 code's specifics are provided in Table 3.

TABLE 3. SPECIFICS OF MAAP5 CODE

GENERAL INFORMATION	
Code name including acronym (and current version)	Modular Accident Analysis Program (MAAP) PWR and BWR: version 5.04, 5.05 beta CANDU: version 5.00a VVER: 5.03 beta
Developing organization	Electric Power Research Institute (EPRI)
Severe accident applicability (e.g. BWR, PWR, PHWR, spent fuel pool, dry storage, etc.)	BWR, PWR, PHWR, VVER, BWR spent fuel pool, PWR spent fuel pool.
Describe the capability for user defined functions transients available? (e.g. Control functions, user defined logic functions, etc.)	User defined functions in the input and capability in BWR and PWR Windows versions to link to user defined dynamic link libraries (external code).
What is the nodalization approach of the code? (e.g. free nodalization, pre-defined nodalization)	RCS: pre-defined Containment: free
Capability to interface with special codes? (e.g. gas flow, CFD, atmospheric dispersion, Origen, etc.)	Limited
Which code features allow users to conduct/explore uncertainty and sensitivity analysis? If yes, is this functionality integrated or separate (e.g. integrated, external tools such as Dakota, SUSA, SUNSET, RAVEN, etc.)	Separate codes must be used.
SOFTWARE LICENSING INFORMATION	
Software licensing organization	EPRI
Who are the intended users? (if any: regulators, industry, etc.)	Industry, research, and regulators.
Who can receive a license to your code? (organization/cluster/individual/individual computer)	Organization
Is the source code available?	Not generally. Certain organizations can obtain a license to the source by special contracts.
What is the agreement for receiving the code?	Industry: must be a member of EPRI's Risk and Safety Management (RSM) program. Vendors and Simulator Developers: Special contract and must be a member of the user's group.
Please provide a general process for prospective users to apply for the code:	Contact Tom Kindred (tkindred@epri.com)
CPU REQUIREMENTS	
What CPU platform does your code work on (Windows, Mac, Linux)?	Windows and Linux
Compiler	Intel Fortran
High performance computing capabilities	None. HPC clusters can be used for Monte-Carlo studies.
STATUS OF DOCUMENTATION	
Is there any documentation on installation instructions?	Yes, this is delivered with the code
Describe available documentation available to users (e.g. user guides, input requirements, modelling guidance, validation)	Yes, this is delivered with the code. There is also a public applications guide that includes a section on benchmarking.
Is documentation publicly available?	The applications guide is public.

TABLE 3. SPECIFICS OF MAAP5 CODE (cont.)

SUPPORT	
What additional resources are available for users? (e.g. technical support, bug reporting, etc.)	Support is provided through the MAAP Users' Group (MUG), which includes user support and error reporting (quarterly).
Is there a user group for the code?	Yes
Describe any training workshops available for users from the code developer:	Training is provided annually by EPRI and periodically by the development contractors. There is also computer based training for introductory users.
Are test cases or examples available for user validation of their installation?	Yes
Are reference plant models or input decks available to users?	Yes
Briefly describe what future, near term developments will be implemented into the code:	Lessons learned from Fukushima Daiichi NPP RoK passive containment cooling design Accident tolerant fuel properties.
Describe any supporting software for I/O processing or a GUI:	I/O is processed via text files. A plotting package is included with the software. Limited GUI is available for some reactor designs.
Representative Publications	— ELECTRIC POWER RESEARCH INSTITUTE, Modular Accident Analysis Program 5 (MAAP5) Applications Guidance: Desktop Reference for Using MAAP5 Software—Phase 3 Report, EPRI, Palo Alto (2017).

2.2.4. MACCS code

MACCS is a MELCOR Accident Consequence Code System used for calculating health and economic consequences from a release of radioactive materials into the atmosphere and is the US Nuclear Regulatory Commission code used to estimate the offsite consequences of potential, severe accidents at NPPs. The code is used to perform probabilistic health and economic consequence assessment of hypothetical releases of radioactive material. Atmospheric transport and dispersion, including wet and dry deposition, probabilistic treatment of meteorology, environmental transfer, countermeasure protective action strategies, dosimetry, health effects, and economic impacts can all be evaluated by the code. MACCS is used by domestic and international government agencies and industry for Level 3 PRA analyses. It is also used by the US Department of Energy (DOE) to perform documented safety analyses of their facilities. MACCS calculates consequences for the three phases defined by the US Environmental Protection Agency: the emergency, intermediate, and long term phases.

ATMOS performs all the calculations pertaining to atmospheric transport, including dispersion and deposition of single or multiple source terms, as well as the radioactive decay and ingrowth that occurs prior to release, while the material is in the atmosphere, and after it deposits. The specification of the release characteristics designating a 'source term(s)' can consist of up to 500 plume segments, each representing a constant release rate over some period. The ATMOS models the transport of these plume segments considering time varying meteorological conditions. Single weather sequences can be evaluated or weather variability can be treated via several sampling options. When weather sampling is used, results are reported as statistical summaries and optionally as a complimentary cumulative distribution function (CCDF). In addition to weather variability, uncertainty in other input parameters can be treated as well. In addition to the air and ground concentrations, ATMOS determines plume arrival time, plume departure time, and plume dimensions. These plume features are used to determine whether evacuees interact with the plume segments as they travel through the grid.

The EARLY module models the period starting with accident initiation. This period is commonly referred to as the emergency phase. It may extend up to 40 days after the arrival of the first plume segment at any downwind location. In the EARLY module the user may specify emergency response scenarios that include sheltering, evacuation, and dose dependent relocation. The EARLY module has the capability to combine results from one to twenty different emergency response cohorts, which are used to define different behaviours within the population. EARLY radiation doses consider five pathways:

- Direct external exposure to radiation from the plume (cloudshine);
- Exposure from inhalation of radionuclides in the cloud (cloud inhalation);
- Exposure to radioactive material deposited on the ground (groundshine);
- Inhalation of resuspended material (resuspension inhalation);
- Skin dose from radionuclides deposited onto the skin.

The CHRONC module simulates the events that occur following the emergency phase modelled by EARLY. Various long term protective actions can be taken during this period to limit radiation doses to acceptable levels, including interdiction, decontamination, and condemnation of property. CHRONC calculates the individual health effects that result from both external and internal dose pathways. CHRONC also calculates the economic costs of the long term protective actions as well as the cost of the emergency response actions that were modelled in the EARLY module. Three long term exposure pathways are modelled to predict the radiation exposures from accidental radiological releases: groundshine, resuspension inhalation, and ingestion of contaminated food and water.

MACCS has been widely distributed and used throughout the DOE, by the USNRC, by the US nuclear industry, as well as other organizations, including international regulators and industry. The historical reference to the consequence code has been MACCS or MACCS2. The number 2 has now been dropped in favour of a single version number, for example, MACCS 3.10.0.

In 2001, the NRC initiated an effort to create a Windows based interface and framework for performing the consequence analysis. This effort was intended to facilitate creation and modification of input files, reduce the likelihood of user errors, enable evaluation of uncertainties in input parameters, and displace the original batch framework with a Windows-based framework. The result of this development effort is the WinMACCS code. WinMACCS is currently integrated with versions of MACCS, COMIDA2 (a food chain model for MACCS), and Latin Hypercube Sampling (LHS) to perform all required functionality. The original MACCS2 batch framework is preserved; MACCS can still be run in standalone fashion apart from the WinMACCS interface. However, there are significant advantages to the WinMACCS framework for performing consequence analyses for most users.

Figure 2 shows the lineage of accident consequence codes for the USNRC/DOE and major projects in which the codes were used. Post 2008, for simplicity, and for the purposes of this document, the combined functionality of MACCS and WinMACCS is often referred to as MACCS. Table 4 presents specifics of the MACCS code.

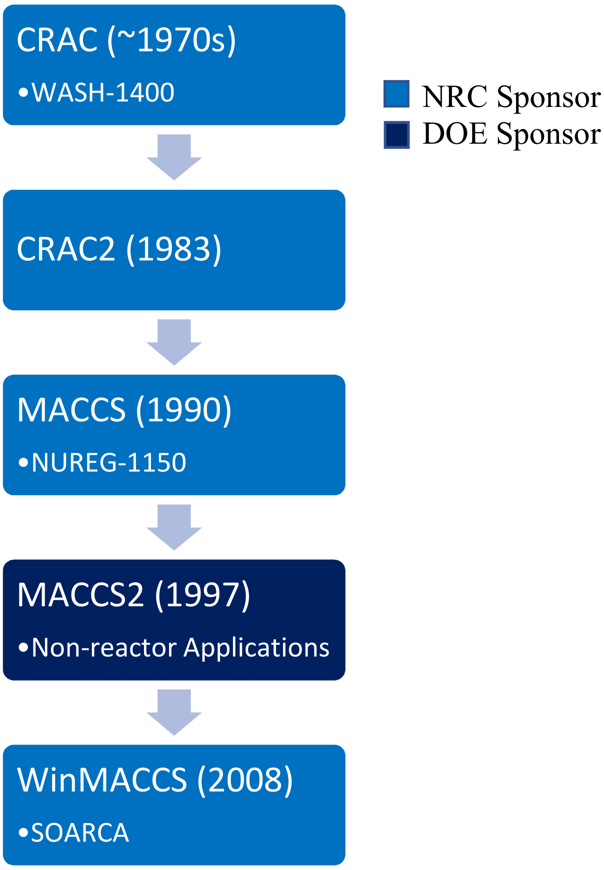

FIG. 2. MACCS code version flow chart.

TABLE 4. SPECIFICS OF MACCS CODE

GENERAL INFORMATION	
Code name including acronym (and current version)	MELCOR Accident Consequence Code System (MACCS) 3.11.2 SVN 5595
Developing organization	Sandia National Laboratories (SNL) under contract to the Nuclear Regulatory Commission
Severe accident applicability (e.g. BWR, PWR, PHWR, spent fuel pool, dry storage, etc.)	BWR, PWR, PHWR, VVER, BWR spent fuel pool, PWR spent fuel pool, any radiological atmospheric dispersion event.
Describe the capability for user defined functions transients available? (e.g. Control functions, user defined logic functions, etc.)	User defined functions in the input and parameter selections.
What is the nodalization approach of the code? (e.g. free nodalization, pre-defined nodalization)	Pre-defined, polar coordinate system.
Capability to interface with special codes? (e.g. gas flow, CFD, atmospheric dispersion, Origen, etc.)	Limited
Which code features allow users to conduct/explore uncertainty and sensitivity analysis? If yes, is this functionality integrated or separate (e.g. integrated, external tools such as Dakota, SUSA, SUNSET, RAVEN, etc.)	Latin Hypercube Sampling (LHS), including an option for simple random sampling, is integrated in WinMACCS.

TABLE 4. SPECIFICS OF MACCS CODE (cont.)

SOFTWARE LICENSING INFORMATION	
Software licensing organization	NRC and SNL
Who are the intended users? (if any: regulators, industry, etc.)	Industry, research, regulators, and international community.
Who can receive a license to your code? (organization/cluster/individual/individual computer)	Any organization within the US, international members of CSARP, international customers licensed by SNL.
Is the source code available?	No
What is the agreement for receiving the code?	Potential user requests code via http://www.nrc.gov/about-nrc/regulatory/research/obtainingcodes.html
Please provide a general process for prospective users to apply for the code:	See maccs.sandia.gov “Get Code” which provides detailed instructions and a link to http://www.nrc.gov/about-nrc/regulatory/research/obtainingcodes.html
CPU REQUIREMENTS	
What CPU platform does your code work on (Windows, Mac, Linux)?	Windows and Linux
Compiler	Fortran 95
High performance computing capabilities	Can run multiple instances in parallel on a single computer with multiple processors or on a cluster using DEF.
STATUS OF DOCUMENTATION	
Is there any documentation on installation instructions?	Yes, included with code.
Describe available documentation available to users (e.g. user guides, input requirements, modelling guidance, validation)	Upon installation, the user gets the following references: — MACCS Users’ Guide; — MACCS Model Description; — Code Manual for MACCS2 Vol. 1 & 2; — NUREG-1935, Parts 1 & 2; — NUREG/CR-6853; — NUREG/CR-7009; — NUREG/CR-7110 Rev. 1, Vol 1 & 2; — NUREG/CR-7161.
Is documentation publicly available?	All documents are publicly available.
SUPPORT	
What additional resources are available for users? (e.g. technical support, bug reporting, etc.)	User support is provided to all MACCS users through the NRC contract. Support is also provided through the International MACCS Users’ Group (IMUG) and the Asian MELCOR/MACCS Users’ Group (AMUG). Bug reporting is available using Bugzilla.
Is there a user group for the code?	Yes
Describe any training workshops available for users from the code developer:	Training is provided annually to NRC staff (P-301) and to the general community via the annual MACCS Users’ Workshop.
Are test cases or examples available for user validation of their installation?	Yes
Are reference plant models or input decks available to users?	Yes
Briefly describe what future, near term developments will be implemented into the code:	New GDP-based economic model and HYSPLIT atmospheric transport model will be in MACCS 4.0, which is to be release in the first half of 2019.
Describe any supporting software for I/O processing or a GUI:	MelMACCS, SecPop, and plume animation software.
Representative Publications	— State–of–the–Art Reactor Consequence Analysis (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analysis (Draft)

2.2.5. MELCOR code

MELCOR is a fully integrated, engineering level computer code developed by Sandia National Laboratories for the USNRC to model the progression of severe accidents in NPPs. Development of MELCOR was motivated by Wash1400, a reactor safety study produced for the NRC, and the Three Mile Island NPP accident. Since the project began in 1982, MELCOR has undergone continuous development to address emerging issues, process new experimental information that emerged following the TMI Unit 2 accident, and has created a repository of knowledge on severe accident phenomena. Mechanistic codes such as CORCON, VANESSA, and CONTAIN have either been integrated into the MELCOR code or effectively replaced by MELCOR as its capabilities have expanded. This leads to an integrated systems level code for performing PRA analyses evaluating the full reactor accident sequence.

MELCOR has an extremely large user base spanning the entire globe. As shown in Figure 3 there are more than 990 licensed MELCOR users in Asia, Europe, the Middle East, and North and South America. Users have organized both a European MELCOR User Group (EMUG) as well as an Asian MELCOR User Group (AMUG) which meets annually and offers an opportunity for code users to share experience in using the code as well as an opportunity to discuss issues with code developers. Annual workshops are also provided for users to gain greater insight and experience in using the code.

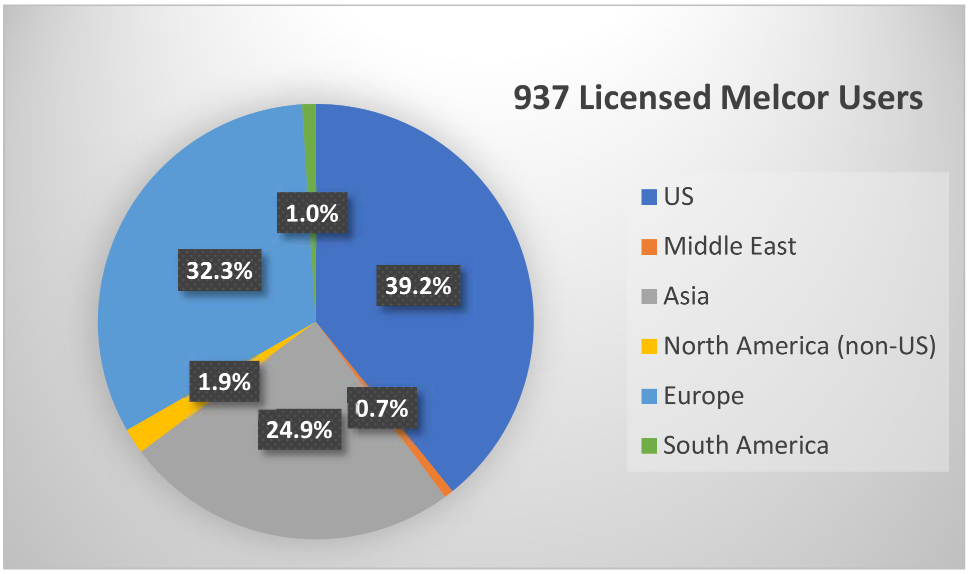

FIG. 3. MELCOR code users [16].

MELCOR models a broad spectrum of severe accident phenomena, both in boiling water and pressurized water reactors, in a unified framework. These phenomena include: thermal hydraulics response in the reactor coolant system, reactor cavity, and the containment and confinement buildings, core heat up, degradation, and relocation, core concrete attack, hydrogen production, transport, and combustion, fission product release and transport behaviour. The modelling is generally simple with the objective of capturing the important physics processes. Physics models are grouped into code packages which explicitly exchange information.

MELCOR utilizes a control volume approach to modelling the thermal hydraulics. Control volumes represent two different fields, 'pool' and 'atmosphere', and assume stratification under gravity. Each phase may have different temperatures but have the same pressure at the interface. In addition, a liquid phase may be suspended in the atmosphere (fog) as well as a vapor phase in the pool (bubbles). Furthermore, non-condensable gases may be present in the atmosphere phase. In addition, there is mass and momentum exchange between the two phases. Control volumes are connected through flow paths which are a construct for determining pressure losses as fluid flows between volumes. Two fluid hydrodynamics with six equations for conservation of mass, energy, and momentum are calculated where v2 terms in the kinetic energy and momentum flux terms are typically ignored. A solution is obtained by linearizing the velocity equation and the method assures a very accurate conservation of mass.

Sandia Labs has taken the approach of integrating the modelling for various reactor types within a single code executable. From the user perspective, this means specifying a reactor type and developing input decks within a familiar syntax for various reactor types. From a developer's perspective, this means designing code in a generalized manner that allows code components to represent different reactor components (depending on reactor type) with characteristics that are dependent on the reactor type being modelled. This also simplifies code maintenance as a large number of routines are common to various reactor types. Currently, models exist for boiling and pressurized reactors, high temperature gas reactors, sodium containment fires, and spent fuel pools. There is also a version of MELCOR that has wide spread application to fusion reactor safety and design.

Though MELCOR's primary application is for estimation of severe accident source terms, and their sensitivities and uncertainties for a variety for regulatory needs it is also applied to leak path factor analysis for non-nuclear facilities.

The MELCOR code specifics are provided in Table 5.

TABLE 5. SPECIFICS OF MELCOR CODE

GENERAL INFORMATION	
Code name including acronym (and current version)	MELCOR 2.2
Developing organization	Sandia National Laboratories
Severe accident applicability (e.g. BWR, PWR, PHWR, spent fuel pool, dry storage, etc.)	PWR, BWR, high temperature gas reactors (PBR and PMR), spent fuel pool, and sodium fires.
Describe the capability for user defined functions transients available? (e.g. Control functions, user defined logic functions, etc.)	Multiple built in control functions (arithmetic, logical, user defined) as well as vectorized control functions.
What is the nodalization approach of the code? (e.g. free nodalization, pre-defined nodalization)	Free nodalization
Capability to interface with special codes? (e.g. gas flow, CFD, atmospheric dispersion, Origen, etc.)	MELCOR generates MACCS output directly. Interface to other codes through explicit coupling only.
Which code features allow users to conduct/explore uncertainty and sensitivity analysis? If yes, is this functionality integrated or separate (e.g. integrated, external tools such as Dakota, SUSA, SUNSET, RAVEN, etc.)	Input fields and sensitivity coefficients can be used for uncertainty analysis. Functionality is separate through custom developed tools as well as Dakota (via SNAP).

TABLE 5. SPECIFICS OF MELCOR CODE (cont.)

SOFTWARE LICENSING INFORMATION	
Software licensing organization	Sandia National Laboratories
Who are the intended users? (if any: regulators, industry, etc.)	Regulators and researchers though also used by industry.
Who can receive a license to your code? (organization/cluster/individual/individual computer)	Individual computer (node locked). User or organizations may request multiple licenses.
Is the source code available?	No
What is the agreement for receiving the code?	Cooperative Severe Accident Research Program (CSARP) and Non-Disclosure Agreement.
Please provide a general process for prospective users to apply for the code:	International governmental organization located in a CSARP member country: — Access is provided through country's CSARP representative. International non-governmental organization located in a CSARP member country: — Access to the code is provided through country's CSARP representative. — In addition, fill out, sign and return non-disclosure agreement. International organization located in a non-member country: — Request the code through the NRC's Office of International Programs.
CPU REQUIREMENTS	
What CPU platform does your code work on (Windows, Mac, Linux)?	Windows, Mac, and Linux
Compiler	Intel 11.1.065
High performance computing capabilities	No, HPC clusters can be used for Monte Carlo studies.
STATUS OF DOCUMENTATION	
Is there any documentation on installation instructions?	License activation documentation. No other documentation on installation.
Describe available documentation available to users (e.g. user guides, input requirements, modelling guidance, validation)	User-guide (syntax), Reference Manual (model descriptions), Validation Manual.
Is documentation publicly available?	Yes
SUPPORT	
What additional resources are available for users? (e.g. technical support, bug reporting, etc.)	Technical support, bug reporting. MELCOR URL: http://energy.sandia.gov/energy/nuclear-energy/nuclear-energy-safety-technologies/melcor/
Is there a user group for the code?	European MELCOR User Group; Asian MELCOR User Group.
Describe any training workshops available for users from the code developer:	Annual training workshop in conjunction with CSARP as well as occasional workshops for user groups.
Are test cases or examples available for user validation of their installation?	Yes
Are reference plant models or input decks available to users?	Not yet
Briefly describe what future, near-term developments will be implemented into the code:	Non-LWR models, Accident Tolerant Fuels
Describe any supporting software for I/O processing or a GUI:	SNAP (GUI and post-processing), PTFREAD (EXCEL Add-in for post-processing), MELCOR Launchpad (front-end for running code), MELCOR syntactical library for NotePad++

TABLE 5. SPECIFICS OF MELCOR CODE (cont.)

SUPPORT	
Representative Publications	— SANDIA NATIONAL LABORATORIES, MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.2.9541, SAND 2017-0455 O, SNL, (2017). — SANDIA NATIONAL LABORATORIES, MELCOR Computer Code Manuals, Vol. 2: Reference Manual, Version 2.2.9541, SAND 2017-0876 O, SNL, (2017). — NUCLEAR REGULATORY COMMISSION, State-of-the-Art Reactor Consequence Analyses Project Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station, NUREG/CR-7155 SAND2012-10702P, U.S. Nuclear Regulatory Commission, (2016). — NUCLEAR REGULATORY COMMISSION, MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project, NUREG/CR-7008, U.S. Nuclear Regulatory Commission, (2014). — NUCLEAR REGULATORY COMMISSION, State-of-the-Art Reactor Consequence Analyses Project, Vol. 1 & 2 (Rev. 1), NUREG/CR-7110, U.S. Nuclear Regulatory Commission, (2013).

2.2.6. RELAP5/SCDAPSIM code

RELAP/SCDAPSIM is a best estimate multi-dimensional, two fluid, non-equilibrium system thermal hydraulics code with options for:

- Detailed fuel and severe accident behaviour models and correlations for LWR/PHWRs;
- Integrated uncertainty analysis;
- Coupled thermal hydraulics 3D reactor kinetics analysis;
- Alternative fluids and correlations for advanced fluid or reactor analysis.

The code is being developed in the framework of the SCDAP development and training program (SDTP) administered by ISS. There are several advanced versions developed by ISS for a wide range of applications and users.

The MOD3.x series of the code is primarily used for applications related to LWR and PHWR designs. RELAP/SCDAPSIM/MOD3.4 is the most widely used of the MOD3.x series. It is recommended for production use since the models and correlations have been frozen after an extensive code assessment period. MOD3.x(RT) has special options for use of the code in full scope training simulators. The accuracy of the current MOD3.x models and correlations is currently being reassessed using an extensive range of historic integral thermal hydraulic/severe accident experiments through the SDTP sponsored university support program. All results of the reassessment will be available in the open literature.

MOD3.5 was developed initially to support the design and analysis of integral thermal hydraulics and severe accident experiments performed in Europe since the mid of 1990s. It is also used to support ongoing Fukushima Daiichi NPP related decommissioning analysis and research and development activities. MOD3.5 is one of the primary design analysis codes used to support the ongoing Quench experiments performed in Germany.

MOD3.6 has special modelling options for PHWRs with vertical and horizontal fuel channels.

The MOD4.x series of the code is the first version of RELAP5 and RELAP/SCDAPSIM completely rewritten to FORTRAN 90/95/2000 standards. MOD4.x also includes advanced numerical options such as improved time advancement algorithms, and improved water property correlations including options for supercritical water applications. Alternative fluids currently in MOD4.x include Na, Pb alloys, and a variety of molten salts such as FLiNaK, FLiBe, KFZrF4. The latest version of the MOD4.x series is MOD4.1 where options to treat the presence of non-condensable gases with liquid metals/salts are being developed and tested.

The specifics of the code are provided in Table 6.

TABLE 6. SPECIFICS OF RELAP5/SCDAPSIM CODE

GENERAL INFORMATION	
Code name including acronym (and current version)	RELAP/SCDAPSIM — a variety of versions are under developed to support a range of applications and versions.
Developing organization	Innovative Systems Software, (ISS) and selected members of the international SDTP cooperative code development program.
Severe accident applicability (e.g. BWR, PWR, PHWR, spent fuel pool, dry storage, etc.)	The code is used for a variety of applications including current and advanced nuclear power plants and research reactors. The code is also used to design and analyse integral thermal hydraulic experiments. The MOD3.5 version has special models and correlations for air ingression and so has been used for spent fuel pool analysis.
Describe the capability for user defined functions transients available? (e.g. Control functions, user defined logic functions, etc.)	The code has an extensive control system capability which allows the users to build complex control logic through input. Some users have also integrated control system routines developed using 3rd parties or by the end users.
What is the nodalization approach of the code? (e.g. free nodalization, pre-defined nodalization)	The code uses a flexible nodalization for hydrodynamic system components, detailed fuel and severe accident components, and porous media. For example, users can build representative fuel assemblies using a variety of detailed SCDAP components including LWR fuel rods, electrically heated fuel rod simulators, control rods/blades, and generalized heat structures. The porous medium is based on a 2D finite element approach so users can include finite element nodalizations developed by 3rd party finite element mesh generators.
Capability to interface with special codes? (e.g. gas flow, CFD, atmospheric dispersion, Origen, etc.)	The code has been interfaced to a variety of other codes and simulator GUI environments using standardized interfaces. Examples included user supplied 3D reactor kinetics packages, subchannel thermal hydraulic codes like COBRA-TF, GRAPE and RELSIM (desktop simulator GUI environments)
Which code features allow users to conduct/explore uncertainty and sensitivity analysis? If yes, is this functionality integrated or separate (e.g. integrated, external tools such as Dakota, SUSA, SUNSET, RAVEN, etc.)	RELAP/SCDAPSIM includes an integrated uncertainty analysis capability. This package can include the influence of (a) input and (b) source code variables not accessible through input. The user identifies the parameters that they want to include and their associated uncertainty distributions. This information along with the best estimate input model then allows the code to compute uncertainty distributions for selected output quantities like peak cladding temperature as a function of time. The code also provides ranking statistics for the parameters as functions of time in spreadsheet compatible format. See the third representative publication for example of the application of the option for the assessment of uncertainties in German Quench-06 integral bundle heating, melting, and quenching experiment. Detailed descriptions of the methodology and other examples of the applications of this option are available in the open literature.

TABLE 6. SPECIFICS OF RELAP5/SCDAPSIM CODE (cont.)

SOFTWARE LICENSING INFORMATION	
Software licensing organization	RELAP/SCDAPSIM is copyrighted by Innovative Systems Software (ISS). It is distributed by ISS as well as international marketing representatives in India and China.
Who are the intended users? (if any: regulators, industry, etc.)	The code is used by regulatory bodies, vendors, utilities, engineering consultants, research organizations, and universities. See the first and second representative publications for overviews of some of the applications of the code by a range of users.
Who can receive a license to your code? (organization/cluster/individual/individual computer)	A variety of licensing options are available for the RELAP/SCDAPIM user community including individual computers and servers. Site licenses are available for selected versions of the code. Temporary and permanent licensing options are available. Compile licenses are available, as noted below, for those users wanting to modify the models and correlations in the code.
Is the source code available?	Yes — a variety of compile license options allow the user to review and modify the source coding for the models and correlations. A full source code license is only available for RELAP/SCDAPSIM/MOD3.4.
What is the agreement for receiving the code?	Standard licensing agreements are available through ISS and selected marketing representatives in India and China.
Please provide a general process for prospective users to apply for the code:	Users can contact ISS through our web page — www.relap.com to obtain detailed licensing options and price quotes. In India and China, users can contact ISS marketing agents directly for licensing options and price quotes. Contact information for marketing agents are available through www.relap.com.
CPU REQUIREMENTS	
What CPU platform does your code work on (Windows, Mac, Linux)?	The code has been installed on machines with Windows, Linux, and Mac OSs.
Compiler	Users wanting to modify the code under the terms of our compile licenses are required to use INTEL supplied compilers for compatibility to our compile libraries. Details about the specific versions of the compiler required are available by request.
High performance computing capabilities	The code can run on typical laptop or desktop computers faster than real time for representative multi-D plant models for a wide range of transients. Multiple CPU desktop computers with large high speed hard drives (T-byte) are recommended when using integrated uncertainty analysis options.
STATUS OF DOCUMENTATION	
Is there any documentation on installation instructions?	Training materials, reference manuals, and sample problems are included in a standard installation file. Installation of the code executable and associated files and running of the sample files typically takes a few minutes. User with compile licenses can also install the code with models and correlations source files and proprietary compile library in a few minutes. Training material on the compilation of the code is included but it is recommended that new users attend one of our regularly scheduled training workshops on code/model development or work with our technical support group to ensure that the user understands how to properly modify and recompile the code. The code/model development training workshops are typically 1 week.
Describe available documentation available to users (e.g. user guides, input requirements, modelling guidance, validation)	The documentation includes detailed descriptions of the (a) modelling theory and equations, (b) constitutive relationships and correlations, (c) user guidelines, (d) input requirements, (e) code assessment, and (f) programming guidelines. Descriptions of ongoing modelling improvements and code assessment activities are published in the open literature on a frequent basis. Supplemental user guidelines, representative input models, and other training material are provided in training workshops.
Is documentation publicly available?	Yes

TABLE 6. SPECIFICS OF RELAP5/SCDAPSIM CODE (cont.)

SUPPORT	
What additional resources are available for users? (e.g. technical support, bug reporting, etc.)	Technical support is provided by staff and consultants located in the United States, Europe, India, Southeast Asia, and China. All-inclusive memberships are available upon request by the users that include access to the different versions of the code, on-call technical support, and a variety of training options for those users that (a) are interested in having access to the latest experimental versions of the code, (b) need special models or correlations added to the code, or (c) just want access to our technical support staff to support their development or analysis activities. Special memberships are available for university under our SDTP university support. The university membership included extended training internships for students or faculty to work with our technical staff.
Is there a user group for the code?	Yes
Describe any training workshops available for users from the code developer:	Training workshops are offered on a frequent basis at different locations internationally.
Are test cases or examples available for user validation of their installation?	Yes, sample problems and training exercises are included with the installation to help the user test the installation as well as learn how to use some of the unique options of the code.
Are reference plant models or input decks available to users?	Yes, representative input models are available for training and testing purposes for users and collaborative members. They include representative LWR/PHWR plant models, representative research reactor models, and a large variety of input models for integral thermal hydraulic/fuel/severe accident experiments.
Briefly describe what future, near-term developments will be implemented into the code:	The development activities are determined by the priorities of the user community and our collaborative members. However, the general development plans for the MOD3.x series include the (a) incorporation of new models and correlations required to support the design and analysis of ongoing separate effects and integral experiments focused on proposed new fuel element materials for LWRs and research reactors, (b) extension of the capabilities of the late phase melt progression models for in-vessel melt retention and decommissioning R&D for Fukushima Daichii, and (c) addition of user options requested by the user community. Descriptions of the ongoing development activities for the MOD3.x series are available in the open literature or available by request. The development of the MOD4.x series focuses on the extension of the models and correlations for advanced fluid or reactor systems. See the fourth and fifth representative publications for more detailed descriptions of the MOD4.x series.
Describe any supporting software for I/O processing or a GUI:	Currently two 3rd party developed GUI environments are offered as standard options for RELAP/SCDAPSIM. GRAPE is developed by Nuclear Engineering Limited in Japan and is intended to support the training of nuclear engineering students and other young engineers involved in the design, analysis, and/or operation of power or research reactors. RELSIM is provided by Risk Management Associates in the United States. A variety of other packages are being developed through collaborative activities through our university support program. They include both advanced GUIs as well as aids to support the development and verification of input models. Descriptions of these options are available in the open literature or available by request.

TABLE 6. SPECIFICS OF RELAP5/SCDAPISM CODE (cont.)

SUPPORT	
Representative Publications	— ALLISON, C.M., HOHORST, J.K., Role of RELAP/SCDAPSIM in nuclear safety, Sci. Tech. Nucl. Install. **2010** (2010). — ANTARIKSAWAN, R., HUDA, Md.Q., et al., "Validation of RELAP/SCDAPSIM/MOD3.4 for Research Reactor Applications", Proc. 13th Intl. Conf. on Nuclear Engineering, Beijing, 2005. — ALLISON, C., LE, B.T., et al., "QUENCH-06 EXPERIMENT POST-TEST CALCULATIONS AND INTEGRATED UNCERTAINTY ANALYSIS WITH RELAP/SCDAPSIM/MOD3.4 AND MOD3.5", Proc. 26th Intl. Conf. on Nuclear Engineering, London, 2018. — ALLISON, C.M., WAGNER, R.J., et al., "The development of RELAP/SCDAPSIM/MOD4.0 for advanced fluid systems design analysis", Proc. 11th Intl. Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety, Gyeongju, 2016. — JIANG, S., PEREZ-FERRAGUT, M., et al., "APPLICATION OF RELAP/SCDAPSIM/MOD4.1 TO THE ANALYSIS OF ADVANCED REACTOR/FLUID SYSTEMS WITH LIQUID MOLTEN SALT IN THE PRESENCE OF NON-CONDENSABLE GASES", Proc. 26th International Conference on Nuclear Engineering, London, 2018.

2.2.7. SOCRAT code

SOCRAT: (System of Codes for Realistic Assessments of Severe Accidents) is a computer code intended for a coupled modelling of a wide range of thermohydraulics, physicochemical, thermomechanical and aerosol processes at all stages of accident progression, starting from initial event and up to corium release following the reactor vessel failure and consequent ex-vessel processes in containment. The code is essentially developed to model VVER NPPs. SOCRAT's field of application includes licensing support, design of safety systems, planning of experiments, PSA support, severe accident management guideline (SAMG) development and verification, crisis centres support, and education. During the Fukushima accident in 2011, SOCRAT was used as one of the numerical tools to support decision making about the need on whether or not to evacuate the Russian population of the Far East.

The SOCRAT code is intended for NPP safety assessment under severe accident conditions. Its development started in late 1990s when three stand-alone codes were coupled in one package called RATEG-SVECHA-HEFEST. This package was intended for safety assessment of new VVER designs. Later its field of applicability was extended to all designs of VVER NPPs, and RATEG-SVECHA-HEFEST was renamed in SOCRAT/V1. Today the version V1 ensures one-through simulation of physical processes at all stages of accident progression: from the initial event to molten corium release from the RPV with account for design features of VVER. Physical and mathematical models and calculation modules of SOCRAT/V1 code provide self-consistent description of a wide range of thermal hydraulic, physical–chemical, and thermal mechanical phenomena at the in-vessel stage of a severe accident. The following software modules are used as components of SOCRAT/V1 code for numerical simulation of severe accidents:

- RATEG, simulating two fluid thermal hydraulics in the circuits;
- SVECHA, simulating physical–chemical processes in the core;
- HEFEST, describing the materials behaviour in the lower plenum and vessel degradation.

RATEG module is intended for simulation of thermal hydraulic behaviour of the primary and secondary circuits. It contains models for different elements such as channels, chambers, pumps, valves, etc., and models for control and instrumentation systems allowing development of the full scale nodalization schemes for complex thermal hydraulic systems. Modelling of the coolant flow in RATEG is realized with a two fluid, two phase hydraulic heterogeneous approach. The coolant is assumed to be in liquid or gaseous phases. Each phase is characterized by its own volume, velocity and temperature and may include several components. For example, liquid phase may contain water and dissolved boric acid or non-condensable gases, and gaseous phase contains steam and non-condensable gases. Interactions of phases (heat and mass transfer, friction) and heat transfer to solid structures depend upon flow regime. The basic thermal hydraulic variables are pressure, void fraction, phasic enthalpies, non-condensable qualities (nitrogen, hydrogen, oxygen), and phasic velocities.

Heat transfer in solid structures (fuel rods, control rods, SG tubes, barrel, shrouds etc.) can be modelled either in one-dimensional or two-dimensional approaches. All heat structures in RATEG module have cylindrical or conic geometry.

SVECHA code package is intended for the modelling of processes of core degradation and allows modelling of the following processes:

- External and internal oxidation of cladding by steam in the steam environment including steam starvation conditions;
- Cracking of oxide layer and enhancement of oxidation rate of cladding;
- Cladding oxide scale reduction in the inert atmosphere up to its complete disappearance;
- Eutectic interaction of UO2 with Zr cladding in solid state;
- Dissolution of UO2 and ZrO2 layer by molten zirconium;
- Oxidation of liquid U–Zr–O mixture and formation of ceramic (U,Zr)O2–x corium during oxidation;
- Change of core configuration due to the relocation of molten materials;
- Formation of blockages during relocation of the melt;
- Failure of fuel elements during accident progression, including FP release;
- Oxidation of steel structures of core;
- Hydrogen release as a result of oxidation reactions;
- Thermal effect of oxidation reactions;
- Heat transfer through the gap between fuel and cladding;
- Radiative heat transfer between cladding, in-vessel structures and reactor walls with account for changes in configuration of the core and in-vessels structures.

Corium behaviour in the lower head after of melt relocation from the core is modelled in HEFEST module. These include interaction of corium material with the structural elements and reactor vessel; heat conductivity in debris; convective heat transfer in a liquid phase; phase transitions (formation and re-melting of crusts); stratification of liquid corium (oxide and metal phases); heat transfer from the melt surface; RPV wall melt-through, melt release from the vessel etc. The core melt is considered as a chemical system: UO2+ZrO2+Zr+steel. In accordance with results of OECD MASCA projects SOCRAT considers that a stratification of metals and oxides may occur in the pool, which can be normal or inverse. Given the design features of VVER (lower plenum with large amount of steel structures and high oxidation degree of corium), a normal stratification is considered with a focusing effect of heat flux from metallic layer on RPV wall breach.

The numerical procedure in HEFEST is based on finite element method solution of a transient, non-linear, 2D energy equation in either axisymmetric or planar calculation domain. The convective heat transfer in the melt is modelled by effective orthotropic coefficients of thermal conductivity that may be spatially non-uniform. It is assumed that the convective flow in the layers is fully developed, so the results of steady-state experiments or CFD based correlations of Nu(Ra) may be applied for estimation of the coefficients.

Validation basis of SOCRAT/V1 consists of both Russian and foreign experimental data of separate physical processes and integral experiments and confirm the modelling adequacy of the processes and phenomena characteristic for severe accident progression in VVER reactors. In 2010 SOCRAT/V1 was licensed by the Russian TSO Scientific and Engineering Centre for Nuclear and Radiation Safety.

Since 2011, the work has concentrated on the development and validation of the advanced version SOCRAT/V3 that additionally allows assessing the radiological consequences of severe accidents. SOCRAT/V3 version is the extension of SOCRAT/V1 code in the field of modelling of radioactive materials buildup in fuel and their behaviour in primary and secondary circuits, and modelling of the thermal hydraulic and physicochemical processes in the core catcher, including the release of radioactive materials (FP).

The following modules have been added to SOCRAT/V3 code to ensure implementation of the above issues:

(a) BONUS — calculates the buildup of FP in the fuel during the irradiation period, and decay heat in fuel after SCRAM.
(b) RELEASE — calculates FP release from the fuel to the gas gap of the fuel rod.
(c) GAPREL — calculates FP release from the gas gap to the primary circuit.
(d) PROFIT — simulates FP behaviour in the primary circuit.
(e) MFPR_MELT — simulates FP release from the molten corium pool in the lower plenum.
(f) RACHIM — calculates the activity, mass and power of isotopes by the given masses of chemical compounds of FP.
(g) HEFEST-EVA — core catcher modelling;
(h) TOCHKA — neutron physical module for calculation of neutron power in the core in point kinetics approximation with account for thermal hydraulic feedbacks including the reactivity insertion, and calculation of decay heat power immediately after SCRAM.

All modules are static libraries that are linked in one executive file. Version V3 includes all modules of V1 version and reproduces fully the functionality of SOCRAT/V1. The input decks developed for SOCRAT/V1 may be directly used with SOCRAT/V3.

The physical models in SOCRAT code have been developed following a reasonable balance between orientation to mechanistic (phenomenological) simulation of physical processes and the use of correlation based models.

For instance, the best estimate phenomenological approach to the modelling is implemented in the models for Zr high temperature oxidation (both solid and liquid phases), candling of molten materials along cylindrical walls, FP release from solid fuel. An important advantage of oxidation modelling in SOCRAT is that the oxidation model is coupled to thermomechanical model that calculates the strains and stresses in the cladding. These data are used to calculate the time when protective zirconia layer starts cracking and the depth of the cracks. Generation of cracks opens a direct way for steam to β–Zr, which intensifies the oxidation process. In turn, the cladding strain and burst is modelled in SOCRAT with account for multiple material layers that develop in the cladding due to phase transformation, oxidation and interaction with fuel pellets. This allows reducing the uncertainty of cladding burst prediction. The module of FP release from solid fuel is based on MFPR code which is a product of collaboration between IBRAE and IRSN. It is coupled to other modules (RATEG, SVECHA, BONUS) and allows calculating the transport of FP across and along the grains to the open porosities, chemical interactions between FP and dissolved oxygen, oxidation of fuel by surrounding atmosphere, and other important processes that determine the rate and composition of FP release from fuel pellets.

In turn, SOCRAT includes several other models that are based on the experimental correlations allowing fast calculations with sufficient accuracy. These are processes of oxidation of stainless steel and B_4C, convective heat transfer in molten pools, different mechanisms of FP deposition, etc.

Separate modules of SOCRAT/V3 code implement numerical modelling of FP and SM transport in the primary and secondary circuits and in the containment. In general, the following physical properties and phenomena are simulated:

— Transport of noble gases, vapor and particles in circuits and in the containment up to the moment of their release to the environment;
— Nucleation;
— Condensation and evaporation in the volume and on wall surfaces;
— Coagulation of particles (gravitational, Brownian and turbulent);
— Deposition of particles due to gravitation, turbophoresis, diffusiophoresis and thermophoresis, diffusion deposition in laminar and turbulent flows, effect of bends.
— Adsorption;
— Gravitational transport of aerosols between calculation cells (intervolume aerosol fallback);
— Deposition of aerosols in the containment by sprinkler system operation;

— Increase of particles due to hygroscopicity.

These physical processes and phenomena are numerically implemented in modules PROFIT (intended for simulation of transport and behaviour of FP and SM in circuits), CONTFP, KIN (both intended for simulation of FP transport and behaviour in the containment). Modules PROFIT, CONTFP, and KIN use similar physical models. Modules CONTFP and KIN are adapted to calculations coupled with standalone lumped parameter containment codes ANGAR and KUPOL, which are connected to SOCRAT through special interfaces. The list of models required for a specific calculation is determined by the user in input deck options.

The HEFEST-EVA module of SOCRAT/V3 code is intended for simulation of processes at the ex-vessel stage of a severe accidents starting from the moment of the corium discharge from the failed reactor pressure vessel, either into the core catcher or onto the concrete floor of the reactor cavity (if the NPP is not equipped with a core catcher).

SOCRAT/V3 was provided with a special interface for coupling with a stand-alone code NOSTRADAMUS that is used for realistic modelling of the atmospheric spread and deposition of radioactive substances, and dose rates to population.

SOCRAT field of application includes licensing support of VVER units, design of safety systems, planning of experiments, PSA2 support, SAMG development and verification, crisis centres support, education.

The specifics of the code are provided in Table 7.

TABLE 7. SPECIFICS OF SOCRAT CODE

GENERAL INFORMATION	
Code name including acronym (and current version)	SOCRAT/V3
Developing organization	IBRAE RAN, «ATOMPROEKT» JSC, «Rosenergoatom» JSC, FSUE RFNC – VNIIEF
Severe accident applicability (e.g. BWR, PWR, PHWR, spent fuel pool, dry storage, etc.)	VVER
Describe the capability for user defined functions transients available? (e.g. Control functions, user defined logic functions, etc.)	Time dependencies of parameters, user defined expressions, setpoints, valves, trips, signals etc. may be simulated with a built-in set of control and logical functions.
What is the nodalization approach of the code? (e.g. free nodalization, pre-defined nodalization)	Free nodalization
Capability to interface with special codes? (e.g. gas flow, CFD, atmospheric dispersion, Origen, etc.)	Interface with NOSTRADAMUS code for the coupled modeling of radiological consequences.
Which code features allow users to conduct/explore uncertainty and sensitivity analysis? If yes, is this functionality integrated or separate (e.g. integrated, external tools such as Dakota, SUSA, SUNSET, RAVEN, etc.)	ELENA, stand-alone module of SOCRAT
SOFTWARE LICENSING INFORMATION	
Software licensing organization	IBRAE RAN, «ATOMPROEKT» JSC, «Rosenergoatom» JSC, FSUE RFNC – VNIIEF
Who are the intended users? (if any: regulators, industry, etc.)	Designers, Research and Educational institutes working on VVER technologies.
Who can receive a license to your code? (organization/cluster/individual/individual computer)	Multiple licenses for organizations working on VVER technologies inside Russian Federation. Outside Russian Federation — currently under discussion.
Is the source code available?	No
What is the agreement for receiving the code?	Currently under discussion
Please provide a general process for prospective users to apply for the code:	Inside Russian Federation — official request should be first submitted to IBRAE RAN. Outside Russian Federation — official request should be submitted to Rosatom organizations which are responsible for the export of VVER technologies.
CPU REQUIREMENTS	
What CPU platform does your code work on (Windows, Mac, Linux)?	Windows, Linux
Compiler	Intel® Visual Fortran Composer XE 2011 Update 11, or Intel® Parallel Studio XE 2018 Update 3
High performance computing capabilities	Supercomputers, HPC clusters
STATUS OF DOCUMENTATION	
Is there any documentation on installation instructions?	Installation instructions are given in User's Guide
Describe available documentation available to users (e.g. user guides, input requirements, modelling guidance, validation)	Description of models and simulation methods, Validation report, User's Guide.
Is documentation publicly available?	No

TABLE 7. SPECIFICS OF SOCRAT CODE (cont.)

SUPPORT	
What additional resources are available for users? (e.g. technical support, bug reporting, etc.)	Technical and methodological support — on a contractual basis. Bug reporting — subversions release and distribution.
Is there a user group for the code?	User group is not formalized, it consists of specialists from different organizations that use SOCRAT in their professional activity.
Describe any training workshops available for users from the code developer:	Trainings within IAEA regional workshops (2012–2018) SOCRAT workshops in Russia (2007–2011).
Are test cases or examples available for user validation of their installation?	Yes
Are reference plant models or input decks available to users?	Yes, simplified samples of VVER-1000 input decks are provided to users.
Briefly describe what future, near term developments will be implemented into the code:	Implementation and validation of high temperature properties for accident tolerant fuel.
Describe any supporting software for I/O processing or a GUI:	RX, SvechaViewer, HefestViewer, SGraph postprocessors
Representative Publications	— NALIVAEV, V., KISELEV, A., LAMY, J.-S., MARGUET, S., SEMISHKIN, V., STUCKERT, J., BALS, Ch., TRAMBAUER, K., YUDINA, T., ZVONAREV, Yu., “The PARAMETER test series”, 3rd European Review Meeting on Severe Accident Research, Nesseber, 2008. — KISELEV, A.E., TARASOV, V.I., TSAUN, S.V., Verification of renovated module for calculation of fission product yield in the framework of integral code SOCRAT, Atom. Energy **113** (2013) 433. — DOLGANOV, K.S., KAPUSTIN, A.V., KISSELEV, A.E., TOMASHCHIK, D.Yu., TSAUN, S.V., YUDINA, T.A., Real-time calculation of the accident at the Fukushima-1 NPP (Japan) using the SOKRAT code, Atom. Energy **114** 3 (2013) 161, 168. — ARUTYUNYAN, R.V., BAKIN, R.I., DOLGANOV, K.S., KISELEV, A.A., TKACHENKO, A.V., TOMASHCHIK, D.Y., TSAUN, S.V., Reconstruction of the North-West radioactive track during the accident at the Fukushima-1 NPP (Japan) using SOKRAT/V3 and PROLOG software, Atom. Energy **116** 3 (2014) 219, 224. — BOLSHOV, L.A., DOLGANOV, K.S., KISELEV, A.E., STRIZHOV, V.F., Results of SOCRAT code development, validation and applications for NPP safety assessment under severe accidents, Nucl. Eng. Des. (under review).

2.3. CONSIDERATIONS FOR THE USE OF SEVERE ACCIDENT CODES

The participants of the Technical Meeting on Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, held in October 2017, included severe accident code developers, experienced code benchmarking users, and code application end users (for example regulatory body members). This spectrum of participants allowed for coverage of the most relevant perspectives of severe accident analysis in detail. One common agreement, from group discussions in all three different topical areas, was the need to improve the code user guidelines for particular applications by emphasizing the physics and chemistry background of the models used to describe the phenomenology of different phases of a severe accident. The aim of this measure is to encourage user awareness of the current limitations of each particular code for simulation of certain phenomena, as well as trying to reduce the negative impact of the user effect, by use of inadequate nodalization, among other issues.

From the presentations and discussion afterwards, a number of considerations for better use of severe accident code results were proposed. The major issues discussed, and the bases of discussion, during the meeting are summarized next.

Severe accident integral codes are a key tool for deterministic analyses of severe accident progression, but the results obtained from different accident scenarios still require careful analysis given the current limitations on models, user effect, input data uncertainty, etc. Severe accident codes are based on one-dimensional control volume and flow path approaches, plus commonly used containment lumped parameter modelling to represent multidimensional phenomena. These approaches limit the applicability of the code for detailed core degradation, debris dynamics, and containment analysis. Thus, it is necessary to first determine if a particular code can be used to simulate severe accident phenomena for a specific plant technology. For containment analyses, such as hydrogen distribution or fission product transport, attempts to generate input data (for example hydrogen source generation/leak rates) for 3D CFD thermal hydraulics codes are being pursued. The direct coupling of severe accident codes to CFD tools is another path being explored. For severe accident consequence and mitigation analysis, source term data will be fed to specialized codes in fission product atmospheric transport dose calculations.

Depending on the intended application of the severe accident code, results may be strongly affected if the user does not follow the specific guidelines of each specific code for proper plant modelling. Examples of this issue are modelling natural circulation phenomena in the lower plenum, spent fuel pool accidents, and passive safety systems. When trying to provide (possibly even by direct coupling) to other special application codes, the user should be aware that the severe accident code yields data from the lumped parameter approach.

Severe accident codes attempt to describe the evolution of an accident; however, the modelling of the physics and chemistry associated to the phenomenology of each of the different sequential phases still has gaps that need more research. For example, degradation rate and slumping prediction timing varies among different codes. As a consequence of this, and from the different clad oxidation models in each code, in-core oxidation and hydrogen generation rates vary also from code to code. After core slumping, the dynamics of debris relocation to the lower head and molten pool configuration further shows noticeable difference among the results from the different codes. Finally, for in-vessel phenomena, thermomechanical considerations for the RPV lower head breach are different for each different code. For the ex-vessel phase of a severe accident, although some models for core catcher systems exist in some codes, the qualification of those models needs to be addressed.

For containment analysis, hydrogen generation from MCCI is another source of uncertainty to the total hydrogen mass and potential risk for deflagration and detonation. Fission product transport, deposition, scrubbing, etc., impact the source term quantification, but models for such phenomena still present some gaps.

Because of these current limitations, carrying out severe accident analysis should follow a well defined methodology of a sequential processes rather than simply performing a series of simulations from a perspective of a deterministic approach. To do this, performing uncertainty and sensitivity analysis, under the framework of a robust methodology, is a key part of the process.

To address the issues just presented, the meeting participants further discussed potential alternatives to diminish, although not necessarily to totally bound, the negative impact of such issues. For example, with the user effect issue, the importance of code training is recognized, particularly directly by the code developer’s team which best understands strategies for proper nodalization and the code limitations. Code training however should be strongly linked to state-of-the-art information of the physical phenomena involved in particular severe accident scenarios. Sensibility and uncertainty analysis best practices should thus be part of code application training.

An assessment of the user effect issue can also be studied via code to code results comparison. The use of available GUI tools is another way to reduce the chance of introducing incorrect data to input decks, and to faster catching errors in nodalization and setting up the accident sequence logic.

The limited knowledge on various phenomena occurring during the evolution of a severe accident can be improved by new and focused experimental programs in areas such as in-vessel melt retention (IVMR), debris dynamics, lower head failure, etc.

For severe accident analysts, sensitivity plus uncertainty analysis should become a common practice, particularly to better understand how uncertainty should be handled in severe accident analysis and severe accident response.

3. OVERVIEW OF TECHNICAL MEETING ON THE STATUS AND EVALUATION OF SEVERE ACCIDENT SIMULATION CODES FOR WATER COOLED REACTORS

Thirty-seven (37) participants from nineteen (19) Member States, together with several IAEA experts, presented the state of the-art simulation codes addressing severe accidents in WCRs and discussed the needs for improvement, identified the gaps and supported the IAEA initiative to launch a new Coordinated Research Project (CRP) on state-of-the-art severe accident analysis. Meeting participants included code developers and end users, coming from the universities, national labs, industry and government organizations.

T. Jevremovic (NENP/NPTDS) served as Scientific Secretary. The meeting was chaired by F. Mascari from ENEA and co-chaired by R. Gauntt from Sandia Laboratory, C. Spengler from Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), P. Wilhelm from Helmholtz-Zentrum Dresden-Rossendorf (HZDR), and T. Hathaway from the U.S. Nuclear Regulatory Commission. The meeting was divided into three technical sessions, each followed by extensive discussions. The technical sessions provided opportunities for participants from code development organizations and end users from national labs, universities and governmental organizations, to share information on their experiences in the use of various severe accident codes. The extensive discussions developed a number of recommendations for future activities.

3.1. SUMMARY OF TECHNICAL SESSION 1: COMPUTER CODES AND MODELS FOR EVALUATION OF SEVERE ACCIDENTS IN WATER COOLED REACTORS

The first Technical Session was dedicated to general considerations of computer codes and models for the prediction of severe accidents in water cooled reactors. The call for papers included the following topics:

- Severe accident modelling phenomena (thermal hydraulic response, heat up, hydrogen generation, nuclear fuel behaviour, fission products release, vessel failure, core melt, molten core–concrete interactions);
- Integrated computer codes in modelling the progression of severe accidents;
- Computer codes to estimate severe accident source terms (in various scenarios and under different conditions);
- Multiphysics code coupling schemes in modelling the progression of severe accidents (including radiological releases);
- Computer codes for modelling full plant responses to severe accident conditions;
- Computer codes for graphical visualization of severe accident progression;
- Limitations and gaps in existing severe accident simulation codes and areas for further development;
- Modelling of the spent fuel pools during severe accidents: current status of capabilities, areas for improvement and benchmark studies;
- Uncertainty and sensitivity analysis of the computer models and codes for evaluation of severe accidents in water cooled reactors;

— Current state of knowledge on the uncertainties and validation and verification of severe accident codes and models;
— Development of new models, and validation and verification of severe accident simulation codes;
— Numerical benchmark studies of the uncertainties in severe accident modelling and simulation codes and methods.

Fifteen (15) papers were submitted for this session, listed in Table 8 and provided on the CD-ROM detailed in Annex II to this TECDOC report. A summary of the presentations and discussion from the workshop participants is provided.

TABLE 8. SUMMARY OF PAPERS AND PRESENTATIONS FOR TECHNICAL SESSION 1: COMPUTER CODES AND MODELS FOR EVALUATION OF SEVERE ACCIDENTS IN WATER COOLED REACTORS.

TITLE	PRESENTED BY
Status of Development of GRS Code System AC^2: Part I: Modelling of Reactor Phenomena	L. Lovasz (Germany)
Status of Development of GRS Code AC^2: Part II: Modelling of Containment Phenomena	C. Spengler (Germany)
Using Deterministic and Probabilistic Methods for MELCOR Severe Accident Uncertainty Analysis	P.D. Mattie (USA)
Severe Accident Modelling and Analysis of VVER 1000 (V-412) for Kudankulam NPP, India	K. Abhishek (India)
Status of MELCOR 2.2 and Plans for MELCOR 3.0	R. Gauntt (USA)
ASTEC Model Development for the Severe Accident Progression in a Generic AP1000	L. Albright (USA)
Ore Degradation Analysis for a Generic German PWR with the Severe Accident Code ATHLET-CD	P. Wilhelm (Germany)
ASTEC, MAAP and MELCOR Benchmark Code Analysis of an Unmitigated SBO Transient in a PWR-900 Like Reactor	F. Mascari (Italy)
Lithuanian Energy Institute Experience in Simulation of Severe Accidents in Water Cooled Reactors	T. Kaliatka (Lithuania)
Status of Development and Applications of SOCRAT Code for Severe Accident Simulation	A. Kiselev (Russian Federation)
Findings from Uncertainty Studies Evaluating Severe Accident Phenomena and Off-site Consequences Using MELCOR and MACCS	A. Hathaway (USA)
Ex-vessel Combustible Gas Generation	A. Bieliauskas (Belgium)
Application of Severe Accident Analyses Codes in Safety Justifications and Regulatory Review for Ukrainian NPPs	D. Gumenyuk (Ukraine)
Using New Versions of Severe Accident Codes for VVER-440/213 Type Nuclear Power Plants	A. Nemes (Hungary)
Overview of the Integral Code ASTEC V2.1 Revision 1	A. Bentaib (France)

After presentations for the first technical session concluded, an in–depth discussion was held by meeting participants. The following observations were agreed upon by meeting participants.

In general, a severe accident code is a fully integrated, engineering level code able to simulate thermal hydraulic response in the different part of the reactor (reactor coolant system, cavity, containment, and confinement buildings), core heat up and degradation/relocation phenomena, core–concrete attack, hydrogen production/transport/combustion and fission product release and transport behaviour. Severe accident codes are validated using several severe accident experiments carried out in recent decades and data on past severe accidents that together form a comprehensive validation matrix. Integral severe accident code validation is carried out continuously to improve the models and correlations implemented in each code. As new experiments are performed and technologies developed, code validation must be reassessed to ensure comprehensive code validation. Some meeting participants observed the need to improve validation matrices to include emergent technologies (e.g. passive safety systems) and physical phenomena (e.g. natural circulation) that were not included in past validation matrices, but are becoming important factors in evaluating plant performance under severe accident conditions for advanced WCR designs.

Participants agreed that integral severe accident codes are a key tool for deterministic analyses of severe accident progression which can be applied to a wide spectrum of water cooled reactor designs — meeting participant presentations featured a mix of PWRs, BWRs, advanced WCRs and spent fuel pool severe accident analyses.

The discussion between code developers and end users led to many observations and suggested activities. End users noted that further development of user guidelines for application of code systems to limit the user effect on analysis results would be beneficial. This includes developer recommendations to help users balance model detail and simulation time, development of generic input decks for various types of nuclear power plants (addressing proprietary restrictions), and further explanation of the relevant physics. In particular, reduction of the user effect on the characteristic events simulated with the codes (core support plate failure, oxidation processes, lower head molten pool phenomena, RPV failure) was prioritized. End users generally expressed that continued development of graphical user interfaces supports ease of input deck creation, postprocessing of simulation results, and also reducing user effect.

Participants also addressed the need for analyst confidence in analysis results. Some participants proposed coupling of severe accident codes with more detailed codes, in order to increase the fidelity of code results, while others proposed development of an accepted methodology for performing uncertainty analysis. It was stressed that it is the responsibility of the user to assess the individual deterministic calculation with regard to the uncertainty in the code results (e.g. hydrogen generation, which is known to exhibit large uncertainty). Ultimately the importance of uncertainty and sensitivity analysis as applied to severe accident modelling was agreed upon. More specifically, meeting participants noted that uncertainty analyses for complex systems have played a central role in many applications supporting nuclear reactor safety analysis. It was identified that the application of uncertainty analyses has a key role in severe accident analyses. The application of uncertainty methodology enhances usefulness and credibility of severe accident modelling. It provides an unbiased representation and assessment of the uncertainty inherent to the application of models which approximate complex physical phenomena.

3.2. SUMMARY OF TECHNICAL SESSION 2: NEW DEVELOPMENTS IN COMPREHENSIVE COMPUTER MODELLING OF THE FUKUSHIMA DAIICHI ACCIDENT

The second Technical Session was dedicated to latest developments in modelling of the Fukushima Daiichi NPP accident. The call for papers included the following topics: computer codes' capabilities in modelling the Fukushima Daiichi NPP accident, benchmark studies of the Fukushima Daiichi NPP accident, uncertainties and sensitivity studies of the computer simulated propagation and mitigation of the Fukushima Daiichi NPP accident.

Five (5) papers were submitted for this session, listed in Table 9 and provided on the CD-ROM detailed in Annex II to this TECDOC report. A summary of the presentations and discussion from the workshop participants is provided.

TABLE 9. SUMMARY OF PAPERS AND PRESENTATIONS FOR TECHNICAL SESSION 2: NEW DEVELOPMENTS IN COMPREHENSIVE COMPUTER MODELLING OF THE FUKUSHIMA DAIICHI ACCIDENT.

TITLE	PRESENTED BY
MELCOR 2.2 Analyses of the Accidents at Fukushima with Emphasis on Source Term and Key Lessons Learned	R. Gauntt (USA)
On the Applicability of Severe Accident Codes as Forensic Tools: A Study on the Unit 1 of the Fukushima Site	C. Lopez (Spain)
Investigation of Performance of Severe Accident Safety Features for Advanced Reactors to Cope with Fukushima Accident and Post Fukushima Requirements	S. Melhem (Jordan)
BWR Mark II LOCA DBA Severe Accident Simulation with RELAP/SCDAPSIM	J. Ortiz-Villafuerte (Mexico)
The Fukushima-Daiichi Accident Computations with ASTEC and How to Match it with the Dose Measurements in the Environment	A. Bentaib (France)

Presentations from the second Technical Session highlighted ongoing efforts to model the Fukushima Daiichi NPP accident at various organizations (SNL, CIEMAT, IRSN) as well as severe accident modelling efforts informed by the post-Fukushima safety requirements and Fukushima Daiichi NPP accident analysis (JAEC, ININ). The Fukushima Daiichi NPP accident reconstruction studies are performed under the OECD/NEA BSAF project. The phase 1 efforts of BSAF were focused on understanding and reconstructing the basic core damage progression sequences while the phase 2 activities were aimed at characterizing the release of fission products from the damaged reactors and transport of radionuclides to the environment. The current status of the Fukushima Daiichi NPP accident analysis is evaluation of the source term.

Participants discussed many aspects important to severe accident modelling and analysis. The Fukushima Daiichi NPP accident reconstruction studies highlighted novel visualization of transport releases and the importance of nodalization decisions and their effects on results (e.g.

stratification of hot gases). The use of severe accident codes for so called forensic analysis of the Fukushima Daiichi NPP accident, in which severe accident codes are ‘encouraged’ to model identified events, was also discussed.

Meeting participants also highlighted general modelling practices. Such practices included the importance of so called ‘sanity checks’ on code results, and also the importance of good phenomenological understanding of code capabilities and limits of applicability by end users.

These studies were carried out by experienced users and illustrated a ‘forensics’ approach, where the studies were highly informed by known boundary conditions from the accident and known phenomenological events were user imposed, as opposed to predictive. The studies highlighted the ability of codes to capture essential accident signatures when informed by known or suspected events such as venting operations, valve seizure events and core slumping events.

Useful insights from these analyses showed that:

- Adequate containment nodalization into three zones was required to capture observed Fukushima Unit 1 pressure trends;
- Subdivided axial nodalization of the BWR wet well was required to capture the observed thermal stratification of the suppression pool;
- Large numbers of calculations were explored to identify plausible scenarios that best replicated observed accident signatures, illustrating the importance of considering uncertainties in severe accident analyses.

The application of severe accident codes to forensics studies such as the Fukushima accident reconstruction studies highlights that many different ‘plausible’ accident scenarios can reasonably capture main observed accident signatures. Best practices in nodalization of the containment response is needed to capture essential signatures and uncertainties in plant boundary conditions need to be considered in rendering realistic span of predicted outcomes. These findings in general should be considered when conducting ‘predictive’ accident analyses.

3.3. SUMMARY OF TECHNICAL SESSION 3: BASIC PRINCIPLE SIMULATORS FOR SEVERE ACCIDENTS

The third and final Technical Session was dedicated to basic principle simulators for severe accidents. The call for papers included the following topics: severe accidents desktop simulators for currently operating WCRs and advanced WCR designs, device independent and/or web based severe accident desktop simulators, benchmark studies and gap analysis in the severe accident desktop simulators for currently operating WCRs and advanced WCR designs.

Six (6) papers were submitted for this session, listed in Table 10 and provided on the CD-ROM detailed in Annex II to this TECDOC report. A summary of the presentations and discussion from the workshop participants is provided.

TABLE 10. SUMMARY OF PAPERS AND PRESENTATIONS FOR TECHNICAL SESSION 3: BASIC PRINCIPLE SIMULATORS FOR SEVERE ACCIDENTS

TITLE	PRESENTED BY
Development of a Fast-Response Guide for Specific Simulations of Severe Accident Scenarios for Trend Analysis, Training and Supporting	S. Mugica (Mexico)
Experience and Competence in Severe Accident Research and Application	Yangxiaoming (China)
The Methodology for Fission Products Release Evaluation for VVER-1000 Under Analytical (Emergency) Center of Russian Regulatory Authority Technical Support	G. Arbaev (Russian Federation)
Multi-Physics Simulator for Core Degradation and Melt Progression in Light Water Reactors During Severe Accidents	R. Sisson (USA)
Simulation of a Station Blackout at the Angra 2 NPP With MELCOR Code	L. Nelbia (Brazil)
SAMG-D: The IAEA Training Toolkit on the Development of Severe Accident Management Guidelines	I. Khamis (IAEA)

Presentations in the third session were focused on development and use of accident diagnosis and decision support tools which encompass a variety of tools ranging from decision support reference documents to severe accident simulators. The principle function of the support reference documents is to allow decision makers, who may not be analysts, the ability to understand the current state of the reactor during severe accidents. One method of support reference document generation presented at the meeting involved performing calculations on selected scenarios informed by PSAs. The library of calculations would then be used by decision makers to understand the current state of the plant. It was noted at the meeting that the creation of such a library of calculations would require simplification of plant models and that the impact of such simplifications on model accuracy would require assessment to ensure the simpler model is still capable of capturing the accident progression. It was also noted that the reduction in accuracy of the overall calculation would be acceptable to the regulator within the framework of trying to understand the decisions of the utility, keeping in mind that the regulatory body is unable to tell the utility how to proceed during the accident.

Several applications of severe accident codes in simulator environment were presented where requirements for fast running code execution were accommodated, for example, by reduced nodalization. The applications illustrated severe accident code use to forecast when major events in an ongoing accident such as when core uncovery, first expected fuel damage and release of radioactivity might be expected. The examples presented explored potential application to severe accident management actions and for emergency arrangements. In these studies, high accuracy is not expected and important trends and potential timing of important events are approximate and intended to support decision making in emergency arrangements and in training activities.

Severe accident simulators were suggested as an alternative method to support decision making through training operators on postulated severe accident scenarios. One methodology was based on faster than real time ASTEC based simulator for estimating evolution of events prior to core damage. After core damage, a library of pre-calculated severe accidents are used to forecast where the accident may be evolving based on data gained in the pre-core damage stage. Another methodology included the ability for a user to interactively alter the conditions to vary the boundary conditions during the calculations. Allowing operators to train on how to handle severe accident scenarios, and possibly to understand the potential impact of decisions in made during the severe accident.

Exploratory work on coupling thermal hydraulics with neutronics and other multi-physics codes was also presented as an approach for training using a basic principle simulator with implications for higher fidelity modelling of accidents such as SBO concurrent with LOCA.

In subsequent discussions meeting participants agreed on the utility of accident diagnosis and other decision support tools. Some participants noted possible uses by regulatory bodies that could monitor and perform analyses to try to understand the decisions made by utilities during severe accidents.

Group discussions also focused on the need for potential users of a severe accident simulator to understand simulator limitations. There is a high confidence in the thermal hydraulic calculations used within simulators, but the transition to the simulation of a severe accident introduces a high degree of uncertainty, and potential users must understand that. Although this is more acceptable for the user who is trying to understand multiple scenario decisions. Generally, it was agreed that severe accident simulators would be valuable tools, and severe accident training would also be helpful, but the uncertainty in the codes must be stressed and understood by those using the tool. Namely minor alterations in decisions and timing can change the boundary conditions of the calculation which could potentially have a drastic impact on the results.

4. SUMMARY AND RECOMMENDATIONS

Accurate prediction of source term and modelling of severe accident progression by severe accident analysis codes is integral to the continued safe operation of water cooled reactors in both developing and developed Member States. Severe accident analyses performed using state-of-the-art severe accident analysis codes are of interest to operating organizations, technical and scientific support organizations and regulators for their ability to evaluate plant performance and response under severe accident conditions. In addition, severe accident analysis codes may be used to inform the progression of severe accidents in the event of their unlikely occurrence. A number of severe accident analysis codes have been developed internationally by different Member States, some common examples include: AC^2 (GRS, Germany), ASTEC (IRSN, France), MAAP (EPRI, USA), MELCOR (SNL, USA) and SOCRAT (IBRAE, Russian Federation).

The primary purpose of severe accident analysis codes is for prediction of the source term released to the environment during severe accidents, which is then utilized by highly specialized atmospheric transport codes to evaluate the transport of radionuclides in the surrounding environment. In recent decades, integral severe accident analysis codes have been developed, allowing analysts to model the progression of severe accidents from the initiating event up through the release of radionuclides to the environment. Severe accident analysis code performances are validated against a validation matrix of past experiments and past severe accidents such as the Three Mile Island accident in the USA, 1979.

Though severe accident codes are all comprehensively validated based on past experiments, there remains disagreement among calculations performed using different codes for the same accident scenario. Such differences and their causes are currently being explored by a number of organizations as well as by international cooperation between code developers through code to code benchmarks. Furthermore, severe accident phenomena model improvements are being made with new insights gained from experiments and with the development of new models to account for previously ignored phenomena.

Due to uncertainties in boundary conditions and the complex interactions being modelled after the onset of core degradation phenomena, a large degree of uncertainty can persist in severe accident analysis code calculations. Uncertainties in severe accident analysis can be treated through the development of accepted sensitivity and uncertainty analysis methodologies, further advancing risk informed approaches for use in severe accident analysis.

The following are a summary of the recommendations that were documented at the conclusion the October 2017 Technical Meeting held on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors (WCRs). Annex I contains a detailed summary of the technical sessions and includes a tabular listing of the recommendations from the October 2017 meeting. In addition to the October 2017 technical meeting a subsequent consultancy meeting was held in May 2018. The May 2018 consultancy meeting comprised of a team of seven experts from six different countries. This section also summarizes the conclusions from the May 2018 consultancy meeting, in which the recommendations from the October 2017 Technical Meeting were reviewed and discussed to narrow down selection of a potential CRP submission. The purpose of both meetings and this section is to provide an identification of key issues in modelling predictions of severe accidents as well as opportunities for global participation in a future research and development project that will help Member States to improve their preparedness for such unlikely events.

The round table discussion at the conclusion of the October 2017 technical meeting produced a common consensus from the member states based on user experience from this work and observations from the papers and presentations covered in the technical programme. A number of technical issues important to accurate predictions of severe accident behaviour were listed as opportunities for advancing the field. Evaluation of the recommendations during the May 2018 consultancy distilled the material into two primary recommendations, (1) development of a collaborative standard problem and/or experimental blind and (2) advancement of risk informed approaches for use in severe accident analysis focusing on the development of sensitivity and uncertainty analysis methodologies.

4.1. COLLABORATIVE STANDARD PROBLEM/EXPERIMENTAL BLIND PREDICTION

The participants suggested the development of a collaborative standard problem and/or experimental blind predictions to conduct a code to code comparison over the most commonly used severe accident codes. The following phenomena were suggested for a standard problem or experimental blind prediction: in-vessel melt retention, lower head failure, debris relocation, core catcher, natural convection in spent fuel pools during a loss of cooling accident, structural integrity assessment of a pressuriser surge line under the SBO, in-vessel hydrogen generation assessment, assessment of the late in-vessel phase of severe accident progression.

The focus of the suggested studies is to assess the current status of the severe accident codes predictability in modelling various phases and phenomena important during a severe accident progression. With the objective of identification of gaps in the code models, code capabilities and phenomena not currently covered by the severe accident codes. Output from this study would be recommendations for severe accidents code improvement. It was noted by the consultancy members that several of these topics were ongoing or planned areas of research with the global severe accident research community.

4.2. ADVANCEMENT OF RISK INFORMED APPROACH FOR SEVERE ACCIDENT ANALYSES

While it has become standard practice to use risk informed approaches for design basis accidents, only recently have the resources become available to conduct probabilistic analysis using complex systems codes in modelling severe accidents. Some uncertainty analyses are in progress and relevant examples are available in the public international scientific technical literature; however, there is no accepted standard best practice approach or guidance when conducting these analyses for the evaluation of severe accident scenarios for nuclear power plants. Several factors contributed to the sense of urgency and need in this area. The consultancy meeting participants noted several concerns including the heterogeneity of available resources globally, recent development of the computational tools needed to conduct these analysis, lack of standard best practices for model validation, the complex interaction between several key sources of uncertainty including intrinsic solution variability, model uncertainty, model approximations, solution stability/convergence criteria and input uncertainty, which are poorly understood within the global community. There is an urgent need among the global members for the development of common quantitative practices for assessing the combined sources of uncertainties effects on severe accident model output and the sensitivity of the models within the severe accident codes to changes in the input parameters. Advancing the field in this vital area is necessary to increase the confidence of the practitioners with the outcome that it will become a common practice in research framework.

The May 2018 consultancy members recommended that the advancement of risk informed approaches for use in severe accident analysis focusing on the development of sensitivity and uncertainty analysis methodologies take precedence over the other recommendations. Training and learning would be combined through a CRP focusing on a challenge problem with the goal of elevating the ability and sophistication of severe accident code users.

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ABBREVIATIONS

AM	Accident management
ASTEC	Accident Source Term Evaluation Code
ATHLET	Analysis of THermalhydraulics of LEaks and Transients
BWR	Boiling water reactor
CCDF	Complimentary cumulative distribution function
CCI	Corium–concrete interaction
CRP	Coordinated research project
CSNI	Committee on the Safety of Nuclear Installations
DCH	Direct containment heating
DOE	United States Department of Energy
ECCS	Emergency core cooling system
EOP	Emergency operating procedure
EPRI	Electric Power Research Institute
FCI	Fuel coolant interaction
FP	Fission product
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH
HWR	Heavy water reactor
HZDR	Helmholtz-Zentrum Dresden-Rossendorf
IAEA	International Atomic Energy Agency
ICTP	Abdus Salam International Centre for Theoretical Physics
IDCOR	Industry Degraded Core Rulemaking
IEM	International Experts' Meeting
LBLOCA	Large break LOCA
LHS	Latin Hypercube Sampling
LOCA	Loss of coolant accident
LWR	Light water reactor
MAAP	Modular Accident Analysis Program
MACCS	MELCOR Accident Consequence Code System
MCCI	Molten core concrete interaction
NEA	Nuclear Energy Agency
NPP	Nuclear power plant
NPTDS	Nuclear Power Technology Development Section, IAEA
OECD	Organisation for Economic Co-operation and Development
PAR	Passive autocatalytic recombiner
PHWR	Pressurized heavy water reactor
PRA	Probabilistic risk assessment
PSA	Probabilistic safety assessment
PWR	Pressurized water reactor
R&D	Research and development
RCS	Reactor coolant system
RPV	Reactor pressure vessel
SAMG	Severe accident management guidelines
SAMG-D	Severe accident management guideline development

SBLOCA	Small break LOCA
SBO	Station blackout
SDTP	SCDAP development and training program
SG	Steam generator
SGTR	Steam generator tube rupture
SOARCA	State-of-the-art reactor consequence analyses
SOCRAT	System of Codes for Realistic Assessments of Severe Accidents
SOT	Start of the transient
SNL	Sandia National Laboratories
STCP	Source Term Code Package
TMI	Three Mile Island
USNRC	United States Nuclear Regulatory Commission
WCR	Water cooled reactor

Annex I
SUMMARY OF TECHNICAL SESSIONS

I–1. INTRODUCTION

The Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors Technical Meeting included three technical sessions consisting of presentations by participants. Each technical session concluded with an in–depth discussion by meeting participants, highlighting any outstanding questions, the most relevant conclusions and guidance for future work. Short summaries of each presentation and group discussion were composed by meeting chairs and a volunteer participant to provide a focused review of meeting events. The summaries are included in the following sections. Section I–5 summarizes the established status and needs of different activity categories, along with recommendations to the IAEA.

I–2. TECHNICAL SESSION 1: COMPUTER CODES AND MODELS FOR EVALUATION OF SEVERE ACCIDENTS IN WATER COOLED REACTORS

The first Technical Session was dedicated to general considerations on computer codes and models for the prediction of severe accidents in water cooled reactors. Fifteen (15) presentations were provided on the status and development of the GRS code AC^2 in relation to the reactor phenomena modeling; status and development of the GRS code AC^2 in relation to containment phenomena modeling; uncertainty analyses approach description; analyses for VVER-1000 KKNPP utilizing the ASTEC code; Status of MELCOR 2.2. code and plans for the MELCOR 3.0; application of ASTEC code for an AP1000-LIKE; application of ATHLET-CD for a generic German PWR; crosswalk through ASTEC, MAAP and MELCOR analyses for a PWR900-LIKE; user experience and simulation of various topics: RBMK-1500, IVMR for BWR, SFP and experiments; SOCRAT code application for SA analysis; Uncertainty study of severe accident and off site consequence by using MELCOR and MACCS2; MAAP5 application for ex-vessel combustible gas generation estimation; experience for Russian type PWRs with MELCOR, ATHLET-CD, COCOSYS, RELAP/SCDAP, LAVA and SFP analysis for BWR; ASTEC V2.1.1, MELCOR 2.2, MAAP 5.03 applications for VVER-440; Overview of the ASTEC V2.1.1 code. The following is a summary of presented papers and discussion.

I–2.1. Status of Development of GRS Code System AC^2 Part I: Modelling of Reactor Phenomena

L. Lovasz, GRS (Germany), provided the first of a two-part description of AC^2 (ATHLET + ATHLET-CD + COCOSYS) — an integral severe accident analysis code system being developed by GRS in Germany, which included an overview on the current status of reactor phenomena modelling in AC^2, accomplished using the ATHLET and ATHLET-CD portions of the AC^2 code system. ATHLET is responsible for thermal hydraulic calculations in the primary and secondary systems and ATHLET-CD handles all core degradation phenomena. A review of modelling capabilities of both the ATHLET and ATHLET-CD modules of the AC^2 code system were presented. Future plans for AC^2 were also discussed including: efforts to allow the simulation of local radiation and core degradation effects, advancements in the lower grid plate failure criterion, fission product release modelling, wall ablation effect, and melt stratification. An example BWR SBO transient was presented as a demonstration of AC^2 functionality

Interest in the oxidation models, core relocation modelling and slumping phenomena in ATHLET-CD was expressed. Additionally, more detailed information about modelling of SFP and passive systems was requested. The GRS code developers provided information on the current code capabilities and future developments, especially for the implementation of elaborated models based on thermo-mechanical considerations for the RPV lower head, as well as the ongoing research related to SFP and code application for passive safety systems. In the discussions it was agreed that a compromise between the level of detail in view of the reactor core modelling and the computational effort has to be considered.

I–2.2. Status of Development of GRS Code System AC^2 Part II: Modelling of Containment Phenomena

The presentation by C. Spengler, GRS (Germany), was the second part of a two part description of AC^2 focusing on the current status of containment phenomena modelling by the COCOSYS portion of the code system. Basic model characteristics and areas of research and development in COCOSYS were both presented. The COCOSYS portion of AC^2 models the relevant containment phenomena. Thermal hydraulic phenomena have reached a sufficient validation status and difficulties in large water pools have been overcome by coupling with CoPool, a fast running CFD tool. Development of new AFP model with consistent treatment of geometrical surfaces is ongoing.

In the discussion it was highlighted that there is a general interest in coupling CFD approaches with lumped parameter system codes to overcome limitations of lump-parameter concept. Without utilizing multi-dimensional models, the use of lumped parameter codes requires well specified guidelines for nodalization, which may be very specific to the considered scenario.

I–2.3. Using Deterministic and Probabilistic Methods for MELCOR Severe Accident Uncertainty Analysis

P. D. Mattie, SNL (USA), presented on the current SNL approach to uncertainty analysis which are important aspects to state-of-the-art severe accident analysis techniques. The goal of the approach is to look at the potential outcomes that can occur based on the range of possible initial conditions. The methodology is a multistep process that is used on current deterministic models. Highlighted requirements of the methodology include identification of uncertainties and definition of their distributions (range and shape), generation of random samples, completion of a stability analysis with convergence testing (temporal, statistical, and numerical). Next a global and individual result analysis is performed including multiple regression analysis and individual realization analysis, after which model and uncertainty revisions are considered and the process is repeated or terminated. The uncertainty and analysis method presented allows qualitative and quantitative measures of the effect model and parameter uncertainties as well as bound on uncertain model predictions.

In the discussions the participants agree of the high benefits of uncertainty/sensitivity analysis methods as given in the presentation. Benefits are that the user would have a strong instrument to assess a deterministic simulation and the impact of uncertain parameters. It was identified that there is a need to converge on requirements for such analysis and appropriate guidance for parameter probability distribution. The drawback is that such methodology requires acceptable calculation time to perform a large amount of simulations. There is a European H2020 project on the way focussing those objectives.

I–2.4. Severe Accident Modelling and Analysis of VVER 1000 (V-412) for Kudankulam NPP, India

A. Kumar, NPCIL (India), reported on an ASTEC ICARE (standalone) severe accident analysis of the KKNPP in India. The presentation included an overview of the salient features and safety systems of KKNPP. The accident scenario considered was an 850mm LOCA with 2 accumulators, SBO. A core catcher analysis was also performed. Modelling difficulties were presented on difficulty modelling both in-vessel and ex-vessel phenomena. An experimental program has been undertaken to address difficulties with both top and bottom flooding for model validation.

The discussion focused on the problems mentioned during the presentation regarding the passive safety systems modelling in ASTEC (CCFL). A link for further discussion was made to the EU H2020 IVMR project, and a general advice for code users was given to keep intense contact with the code developers for specific problems and how to deal with them. The discussion showed also that the reliable models for core catcher systems are needed and the limited availability in SA codes is mostly due to the proprietary rights, including materials and specific design effects.

I–2.5. Status of MELCOR 2.2 and Plans for MELCOR 3.0

R. Gauntt, SNL (USA), provided an overview on the current status of MELCOR 2.2 and future plans for the development of MELCOR 3.0. MELCOR is ‘like a lego set’ which has built in fundamental components to address any reactor design. It is fully integrated engineering level code capable to simulate all major core degradation phenomena. Crosswalk comparisons between severe accident analysis codes were discussed, and how differences in developer assumptions and code implementations can cause discrepancies between severe accident analysis code models and results (e.g. differences in the permeability of debris crust). Plans for MELCOR 3 currently include a layered software design based on C++ with an advantage in numerics. Future plans are considering DOE advanced reactors research and uncertainties analysis.

The user expressed interest in the development of MELCOR and SNAP and the users expressed generally an interest in GUI for building input decks (not only for visualization). It was identified also the need of usage of a 3D approach for specific topics (qualification of equipment with local dose rates). The multi-scale issues were not ranked as highest priority by the presenter. It was also identified that more information exchange in that area is necessary.

I–2.6. ASTEC Model Development for the Severe Accident Progression in a Generic AP1000-LIKE

L. Albright, University of Utah (USA), presented the current status of a new project to characterize the progression of severe accidents and identify key grace periods relevant to core degradation in the Westinghouse AP1000. Brief overviews were provided on relevant research, in-vessel core degradation progression in PWRs, the Westinghouse AP1000 design, and ASTEC V2.1. The analysis included the definition of two models: a standard or 'half vessel' model and a 'full vessel' model, with the intention to observe the effects of a large cavity above the upper core plate on core degradation progression. The models were evaluated by a scenario involving loss of flow to the RPV with no AMM leading to core degradation and failure of the RPV lower head. Preliminary results indicate the ability of the model to simulate core degradation.

The discussion showed that specific input data are needed for the building of the model which are not easy to be obtained by open literature. Specific questions were related to the special modelling of the upper head of the reactor with CESAR and ICARE and the differences in the modelling for the both models. Interest was shown by the ASTEC code developers and for future collaboration on that topic was proposed. For example, modelling issues were brought forward for different type of material for implementation in the code.

I–2.7. Core Degradation Analysis for a Generic German PWR with the Severe Accident Code ATHLET-CD

P. Wilhelm, HZDR (Germany), presented work performed on the development of a model of a generic KONVOI type PWR in ATHLET-CD. The model was mainly developed in the frame of the WASA-BOSS project. The model has been applied to two accident scenarios types: SBO and SBLOCA. Results presented pertained to a hypothetical SBO case and demonstrated the model's ability to simulate accidents with core degradation up to the point of RPV failure. The analysis focused on FP releases and included cases for varied core burnup. An overview of current, applicable ATHLET-CD modelling capabilities and a description of the model definition were provided along with a demonstration of the visualization capabilities of ATLAS.

Interest was shown for the simulation time of ATHLET-CD, which actually depends on the scenario under consideration. Generally, it can vary from half a day to several weeks. Interest was shown also for the way of modelling of the different burn up with ATHLET-CD, as the focus was on the application of the OREST/FIPISO modules in integrated manner in the code, where OREST calculates the initial fission product and actinide inventories based on the core configuration, fuel enrichment, burn-up and power history, masses, power and activities (decay heat) and the FIPISO module calculates the time evolution of the fission product and actinides inventory (masses, activities and decay heat) after SCRAM. Interest was shown also in the usage of ATLAS (GUI) for online visualization of ATHLET-CD code results.

I–2.8. ASTEC, MAAP and MELCOR Benchmark Code Analysis of an Unmitigated SBO Transient in a PWR-900 Like Reactor

F. Mascari, ENEA (Italy), reported the results of a study to benchmark ASTEC with MAAP and MELCOR. A simple reactor transient was modelled simulating an unmitigated SBO in a PWR-900 LIKE in each of the three codes. Nodalizations of the PWR-900 were made to be as close as possible. The results between the three codes were in qualitative agreement with quantitative differences. It was commented that MELCOR produced the greatest amount of hydrogen and had a greater convection energy than either MAAP or ASTEC. The most relevant differences between the three codes were observed in the in-vessel hydrogen production prediction, as well as the slumping predictions. Though quantitative differences in the results exist, the agreement in qualitative predictions between the three codes can be used as confirmation of the transient phenomenological evolution of the accident. A strict congruence analysis between the core structure nodalization is endorsed.

Together with questions for the applied modelling regarding the steady state simulation and applied models for oxidation, participants agree that predicted times are expected with a certain deviation for characteristic events (like RPV failure time); user should ensure that the applied models are used within the validity range of the model itself.

I–2.9. Lithuanian Energy Institute Experience in Simulation of Severe Accidents in Water Cooled Reactors

T. Kaliatka, LEI (Lithuania), summarized three severe accident analyses performed by LEI: integral analysis of LBLOCA in RBMK-1500, IVMR in a BWR, a spent fuel pool analysis, and severe accident experiments. performed. The presentation included an overview of the current state of Nuclear power and regulation in Lithuania. The integral analysis described was performed using the RELAP5, RELAP/SCDAPSIM, ASTEC, and COCOSYS codes. Results from the IVMR study in a BWR included an analysis by each of: RELAP/SCDAPSIM, ASTEC, and ATHLET-CD. Presented severe accident experiment analyses were performed using ASTEC and RELAP/SCDAPSIM and included the Phébus and QUENCH tests.

The discussion showed that the availability of different oxidation models in the different codes is an additional source of uncertainty. It was highlighted the need of converging on single reliable model to be available in all codes. Specification of better models for relocation to the lower head, molten pool configuration and prediction of RPV failure is needed. An emphasize was given on problems for modelling SFP at ambient conditions.

I–2.10. Status of Development and Applications of SOCRAT Code for Severe Accident Simulation

A. Kiselev, IBRAE (Russian Federation), reported on the status of the development and applications of the SOCRAT code. The presentation included current capabilities of the code and validation status, as well as a description of its applications. Results were reported from the SOCRAT analysis performed as part of the BSAF project on Fukushima Daiichi NPP units 1, 2, and 3. Results from the FUMAC and ATLAS projects were also reported

Applicability of the code for simulation of BWRs was of special interest in the discussion. The presenter underlined that same physical phenomena in PWR and BWR are covered by the available models in the code. The same model is used for the core catcher and molten pool formation in the lower head. Additionally, more information was requested about the code capabilities for modelling of SFP. With the current approaches applied in system codes, steps towards model application were undertaken.

I–2.11. Findings from Uncertainty Studies in Evaluating Severe Accident Phenomena and Off-site Consequences Using MELCOR and MACCS

A. Hathaway, USNRC (USA), reported on the status of SOARCA, a project to identify the realistic outcomes of severe accidents. The results of the analyses performed on Peach Bottom, Surry, and Sequoyah NPPs were presented and uncertainty analysis was used to rank parameters and the effect of their uncertainty analysis results. In each of the three cases "essentially zero" absolute early fatality risk was found. The peach bottom study found that the public health consequences are smaller than earlier projections and that delayed releases of FPs allowed more time for emergency response. The Surry NPP analysis found slightly smaller releases than original calculations indicated and also that ~10% of the cases involved a SGTR (with both a pressure and thermal element involved in the rupture). Similarly, the Sequoyah plant analysis which simulated a seismic event concluded that FP releases are smaller than those found in previous studies and that the long term risk dominates the health effects because of evacuation.

Interest was expressed on hydrogen issues, and the discussion was related to the effect of the in-vessel hydrogen production and its contribution to the risk assessment of the containment.

I–2.12. Ex-Vessel Combustible Gas Generation

A. Bieliauskas, Westinghouse Electric Belgium, reported on passive autocatalytic recombiner (PAR) sizing criterion at Westinghouse. The presentation included an overview of combustible gas generation outside of the RPV and CO sources and recombination. A sensitivity analysis performed indicated that the water fraction in concrete composition strongly impacts H_2 and CO generation. Similarly, rebar density shows an influence on H_2 and CO production.

It was as positive identified the application of the PAR assessment for ex-vessel conditions, during which the in-vessel produced hydrogen was assumed as already burnt. Interest has aroused in the MAAP capabilities to detect DDT, for which an indication is available in the recent version of MAAP. CO effect (toxicity) during recombination was identified as important issue.

I–2.13. Application of Severe Accident Analyses Codes in Safety Justifications and Regulatory Review for Ukrainian NPPs

D. Gumenyuk, SSTC NRS (Ukraine), presented an overview of the Ukrainian nuclear regulatory body (SSTC NRS) and current activities on severe accident analysis for both the VVER 440 and VVER 1000 in Ukraine including L2PSA, SAMG-D and post-Fukushima Daiichi NPP measures. Code use in Ukraine for severe accident analysis includes MELCOR, ATHLET-CD, COCOSYS, RELAP/SCADAP and LAVA. Analysis conclusions are that the VVER 1000 containment fails in most cases of corium injection into containment, that extra measures are needed to prevent H_2 detonation, and that under pressure of the VVER 1000 containment is possible. Current code limitations were highlighted by a desire by a desire to model two ‘cores’ (the progression of a severe accident in both a reactor and spent fuel pool). Future work is planned on the development of VVER 440 and VVER 1000 models in MELCOR 2.1 and a VVER 440 model in ATHLET-CD/COCOSYS. Validation of plant specific models is also planned.

Interest about additional information about the status of implementation of IVMR as SAMM in VVER440 and the applied modelling for the lower head semi-elliptical geometry of the bottom was shown in the discussion. Consideration about the different code capabilities for representation of such specific geometry was identified. The wish expressed by the presenter for parallel calculation of reactor core degradation and accident in the SFP was not shared by the code developers in view of the very high efforts to implement such modelling compared to the benefits for safety evaluation.

I–2.14. Using New Versions of Severe Accident Codes for VVER-440/213 Type Nuclear Power Plants

The main focus of the presentation by A. Nemes, NUBIKI (Hungary), was the results of a severe accident analysis performed by NUBIKI on the Pak NPP in Hungary (VVER 440) using MELCOR 2.2, ASTEC V2.1.1, and MAAP5.03 Beta. An overview of NUBIKI responsibilities and activities were presented as well as an overview of the Paks NPP. Introductions to each of the three codes were given and differences between western and eastern PWRs that make modelling of eastern PWRs difficult with the current severe accident analysis codes were highlighted. The selected accident scenario was an SBO. User experience on the use three codes was also presented including simplicity of code use and its effects on model development, user preference on code flexibility, and the benefits of a GUI for new users vs. old users.

The applicability of MELCOR for VVER440 analysis was discussed. Considerations were also expressed for the modelling of the release of fission products specifically Sr in the ASTEC code. Regarding the last point code deficiencies are already identified by the code developers. Discussions showed that in MAAP due to the objectives of the code to be a transparent tool user effect on the model adaptation is limited. Other codes provide such flexibility.

I–2.15. Overview of the Integral Code ASTEC V2.1 Revision 1

A. Bentaib, IRSN (France), reported on the current status of the integral code ASTEC V2.1 as well as ongoing efforts at IRSN for its further validation and development. The main development goal of ASTEC is to become a fast running code for use in accident management studies, to have the capability to support experiments and emergency response and to account for the effect of safety systems. Changes to the ICARE module, modelling of iodine chemistry and FP behaviour in SOPHAEROS and FP behaviour in containment were also reported. ASTEC has a four tier validation approach: (1) separate effects tests, (2) coupled effects tests, (3) integral tests, (4) simulation of full plants. The current validation matrix has more than 160 experiments and each major code release is validated by a sub-matrix. The current validation status presented is that there is good agreement with experiments in current containment models, however, quench modelling does not currently show good agreement with experimental results.

I–2.16. Technical Session 1 discussion

The discussion highlighted again that there is a strong wish of users' side for reduction of the computational timing. The code developers stressed that short computation times are feasible for simplified input decks which may be appropriate for particular investigation problem. However, for other considerations more detailed input decks are necessary. Code developers rank code robustness with higher priority compared to fast running capabilities.

The summary discussion of this session is as follows:

(a) It is consolidated that severe accident integral codes are a key tool for deterministic analyses of severe accident progression. Severe accident codes for evaluation of the plant response in are being developed and applied by different organizations. In particular, AC^2 is developed by GRS in Germany, ASTEC code is developed by IRSN in France, MAAP code is developed by EPRI in USA, MELCOR is developed by SANDIA for USNRC in USA, SOCRAT is developed by IBRAE in the Russian Federation.
(b) In relation to the code-user application, several contributions presented the wide spectrum of code application for different reactor designs as for example generic PWR-900, VVER 1000(V-412), AP1000, Generic German PWR, RBMK-1500, BWR, SFP, etc. Along the application of the code, the users underline the influence of the user effect. The application of the severe accident code is not limited to national research program but also to several cooperation platforms as underlined in the presentation (EU-FP, IAEA, OECD/NEA, etc.). The hydrogen generation due to the core degradation processes is still a parameter that shows higher uncertainty. The use of the graphical user interface is confirmed as a tool supporting the users to compile an input deck, applied also in post processing of the simulation results and finally reducing the user effect.
(c) In the last decades several experimental activities were carried out in order to validate the different codes regarding models and correlations for the prediction of severe accident phenomena and processes. Though all codes show a comprehensive validation matrix, area for improvement was identified.

(d) Continuous validation of the codes is in progress in order to improve the current model/correlations and as new outcomes from experiments or nuclear accidents (Fukushima) are available further validation will be performed. It is also to underline further validation of severe accident code for the simulation of passive systems for advanced reactors.
(e) The uncertainty analyses for complex systems have successfully played a central role in many applications supporting nuclear reactor safety analysis. It was identified that the application of uncertainty analyses has a key role in severe accident analyses. The application of uncertainty methodology enhances usefulness and credibility of severe accident modelling. It provides an unbiased representation and assessment of the uncertainty inherent to the application of models which approximate complex physical phenomena.
(f) Several discussions were devoted to the potential of coupling of severe accident codes with more detailed codes, in order to increase the fidelity of the code.
(g) In general, a severe accident code is a fully integrated, engineering level code able to simulate thermal hydraulic response in the different part of the reactor (reactor coolant system, cavity, containment, and confinement buildings), core heat-up and degradation/relocation phenomena, core–concrete attack, hydrogen production/ transport/combustion and fission product release and transport behavior. The discussion of the individual presentations leads to identification of the following main issues:
    - Elaboration of user guidelines for application of code systems (minimum recommendation, explanation also on the physics behind);
    - The responsibility of the user to assess the individual deterministic calculation with regard to the uncertainty in the code results;
    - Reduction of the user effect on the characteristic events simulated with the codes (core support plate failure, oxidation processes, lower head molten pool phenomena, RPV failure);
    - Awareness of the user to make compromises between level of detailed input deck and reduced simulation time (guidelines from developers);
    - Availability of generic input decks for various types of plants;
    - Proprietary issues for development of an input decks/models;
    - Methodology needed for performing uncertainty analysis;
    - New challenges for the codes to cover advanced reactor systems (long TH processes).

I–3. TECHNICAL SESSION 2: NEW DEVELOPMENTS IN COMPREHENSIVE COMPUTER MODELLING OF THE FUKUSHIMA DAIICHI ACCIDENT

The morning sessions were focused on specific applications of TH and severe accident codes where SNL, CIEMAT and IRSN presented contemporary analysis results for the accident reconstruction studies performed under the OECD/NEA BSAF project. The phase 1 efforts of BSAF were focused on understanding and reconstructing the basic core damage progression sequences while the phase 2 activities were aimed at characterizing the release of fission products from the damaged reactors and transport of radionuclides to the environment. Activities presented by Jordan highlighted the specification of post Fukushima safety requirements on the planned Russian reactor. Focused analyses were presented using RELAP/SCDAPSIM on the performance of the mostly passive heat rejection system for moving heat from the horizontal steam generators to an air-cooled system outside of containment. Mexico presented other BWR Mark II analyses of unmitigated LOCA analyses where sometimes difficult to understand results are encountered highlighting the importance of good phenomenological understanding of code capabilities and limits of applicability. The following is a summary of presented papers and discussion.

I–3.1. MELCOR 2.2 Analysis of the Accidents at Fukushima with Emphasis on Source Term and Key Lessons Learned

R. Gauntt, SNL (USA), provided some background on the OECD/NEA BSAF Phase I and Phase II projects was given and an explanation of the site damage and damage states of the Fukushima Daiichi NPP reactors and the initial conditions for each reactor. An overview of each reactor accident progression was give where it was explained that the analyses were performed in a 'forensics' mode where suspected major events based on data observations were imposed on the code progression analysis to better replicate observed data. In Unit 1 a proposed steam inerting and de-inerting scenario was given to explain the timing of the hydrogen explosion. Unit 2 analysis highlighted the extended operation of the RCIC steam driven system, emphasizing the potential use of this system outside of its expected operational envelope. Unit 3 analysis highlighted the three peaks portion of the observed PCV pressures and that considerable expert analysis and interpretation was currently focused on this. Three source terms for each plant were superimposed to conduct a MACCS analysis and ground contamination was estimated.

I–3.2. On the Applicability of Severe Accident Codes as Forensic Tools: A Study on the Unit 1 of the Fukushima Site

C. Lopez, CIEMAT (SPAIN), presents the latest BSAF results for CIEMAT MELCOR analysis of the Fukushima unit 1 accident. It was emphasized that there were sparse data for which to try and predict using the code which required some fitting of parameters such as leaks. With no clearly unique combination of assumptions, there are a number of plausible scenario interpretations. Like the SNL analysis, they extended the analysis to 3 weeks. The nodalization of the containment and primary system are presented where the suppression pool was azimuthally divided and the location of assumed leaks where one leak is assumed from the torus at the bellows and another leak at the PCV head closure location. SOARCA guidelines were used to help in the definition of the CIEMAT model. The fission product behaviour is described where iodine is trapped well in the wet well but a lot of CsM is retained in the RPV. CIEMAT made an analysis of the observed dose rate measurements of the CAMS and compared to synthesized dose rates based on code analysis. Pool scrubbing was also evaluated with scrubbing efficiencies of about 30% over the 3-week period.

Some of the questions were focused on how different users selected their nodalization. Users generally consider their code when defining their nodalization. Claudia noted that the nodalization permitted the capture of some hot gas stratification effects. SNL needed at least 3 axial zones in the PCV to capture well the pressurization of the modelled MSL break.

I–3.3. Investigation of Performance of Severe Accident Safety Features for Advanced Reactors to Cope with Fukushima Accident and Post Fukushima Requirements

S. Melhem, JAEC (Jordan), presented the post-Fukushima requirements that are considered against the Russian AES-92 (Gen III+) design. In lights of potential severe external events aspects of the design such a redundancy and passive system performance are considered. Preservation of ultimate heat sink in severe external events is a priority and can involve performance of both active and passive systems. Hydrogen management by PARS is required to avoid deflagrations or detonations from severe accident hydrogen. Non-permanent or portable equipment is also discussed such as generators and heat exchangers. A passive heat removal system for rejecting steam energy from the steam generators is detailed for the plant design. This steam energy is transferred to air outside of the containment. RELAP-5 analyses were done in confirmation of plant response including the response of the passive steam heat and the effect of external environmental temperatures. RELAP-5 confirmatory analyses are reviewed in detail, showing success of the passive heat removal system.

Questions centred around the functioning of the horizontal steam generators, which are in natural circulation, and the potential effects of main coolant pump seal leaks which can eventually result in loss of water inventory after a long time.

I–3.4. BWR Mark II LOCA DBA Severe Accident Simulation with RELAP/SCDAPSIM

J. Ortiz-Villafuerte, ININ (Mexico), presented on Mexico’s two unit site at Laguna Verde of BWR/5 Mark II design that are both up for license renewal. Some background was presented on the status of the plant PSAs and power up rates. ININ and CNSNS (regulator) are using RELAP/SCDAP and MELCOR. GASFLOW is used for hydrogen behaviour analysis. They have done work in very detailed modelling of the RCS jet pumps to model 10 pumps per loop and find that it makes a difference. They have also modelled in great detail the steam line paths and bends, etc in order to better represent the operation of things like the RCIC system. The RSS code was used to assess melt accumulation in the lower head but the implications of the CRD penetration are perhaps not well accounted for. A fast developing LBLOCA with no injection is highlighted showing the melting of the core and a sudden drop of lots of molten material into the lower plenum. He points out that the instruments level indicators can be out of calibration and require correction in order to really know the water level. The unmitigated accident was reviewed first, and then some mitigations of HPCS injection were reviewed. Injection seems to mitigate the high temperatures that were attained from the unmitigated case. A particularly large oxidation energy release was highlighted that seemed to produce very high fuel temperatures above 4000 K.

Questions for this presentation: Ukraine representative noted that the oxidation energy transient was potentially was repaired in a later version of RELAP/SCDAPSIM. It was suggested that the transient was associated with a quenching model that was intended to better capture the results of a KIT Quench experiment, but that it perhaps was not appropriate for actual reactor application and that a subsequent code update corrected this problem.

I–3.5. The Fukushima-Daiichi Accident Computations with ASTEC and How to Match it with the Dose Measurements in the Environment

A. Bentaib, IRSN (France), presented on analyses with ASTEC conducted using V2.1 for the Phase 1 BSAF effort and a later version of ASTEC for the BSAF Phase 2. The complexity of the BWR fuel assemblies was noted and the fact that ASTEC required some modifications to capture the BWR specific geometry aspects. IRSN has produced a similar analysis of the Unit 1 accident progression. ASTEC also predicts dry well liner penetration by MCCI at about the time of the observed containment depressurization, similar to the SNL MELCOR finding. The Unit 2 analysis is also quite successful in replicating the observed data. Regarding the Unit 2 analysis, it is shown that the assumed water injection flow following the manual depressurization is fairly key as to whether lower head failure results. The second phase of the Fukushima analyses focused on source term predictions using the newest version of ASTEC. The three week analysis of fission product release was presented as well and compared to reverse analyses conducted by Kataka. Using their source term a dispersion analysis using C3X was performed showing good comparison with the available data. Plans for subsequent improved modelling and analyses were detailed.

There was some discussion about the atmospheric transport modelling where it was explained that actually three models are used for near field, far field and global scale.

I–3.6. Technical Session 2 discussion

The summary discussion of this session is as follows:

(a) The session highlighted ongoing efforts to model the Fukushima Daiichi NPP accident at various organizations (SNL, CIEMAT, IRSN) as well as severe accident modelling efforts informed by the Fukushima Daiichi NPP accident analysis (JAEC, ININ). The current status of the Fukushima Daiichi NPP accident analysis is evaluation of the source term. So-called forensic analysis of the Fukushima Daiichi NPP accident using MELCOR was discussed, in which MELCOR is ‘encouraged’ to model identified events.

(b) Nodalization decisions and their effects on results were highlighted during the discussions on the Fukushima Daiichi NPP accident modelling. The example given involved initial nodalization schemes that were unable to capture the stratification of hot gases.

(c) The importance of performing so called sanity checks on code results were also highlighted in the presentation by Mexico and the proceeding discussion. Each severe accident code has its own realm of applicability and users must ensure that their models are within this realm as well as check to make sure that the results are reasonable.

(d) The discussion surrounding the IRSN presentation focused on the dispersion modelling used to create a visualization of the transport of releases resulting from the Fukushima Daiichi NPP accident.

I–4. TECHNICAL SESSION 3: BASIC PRINCIPLE SIMULATORS FOR SEVERE ACCIDENTS

The afternoon sessions were focused on development and use of accident diagnosis and decision support tools which encompass a variety of tools ranging from severe accident simulators to decision support reference documents. Mexico highlighted a simulator tool that allows for interactive user/operator involvement where actions can be initiated in a running calculation. Robust input decks were prepared for Laguna Verde based on the plant PSA. China described several accident simulators used in training activities. The Russian Federation described a methodology based on faster than real time ASTEC based simulator for estimating evolution of events prior to core damage. After core damage a library of pre-calculated severe accidents are used to forecast where the accident may be evolving based on data gained in the pre-core damage stage. Exploratory work on coupling thermal hydraulics with neutronics and other multi-physics codes was presented as an approach for higher fidelity modelling of accidents such as SBO concurrent with LOCA.

The third Technical Session was dedicated to basic principle simulators for severe accidents. Four (4) presentations were provided. Three presentations focused on the development of tools which can be used to understand decisions made during severe accidents or provide indication of how an accident may progress or produce fast results for analyses: Development of a Fast-Response Guide for Specific Simulations of Severe Accident Scenarios for Trend Analysis, Training, and Supporting; Experience and Competence in Severe Accident Research and Applications; and The Methodology for Fission Products Release Evaluation for VVER-1000 Under Analytical (Emergency) Center of Russian Regulatory Authority Technical Support; and Multi-physics Simulator for Core Degradation and Melt Progression in Light Water Reactors During Severe Accidents. The following is a summary of presented papers and discussion.

I–4.1. Development of a Fast-Response Guide for Specific Simulations of Severe Accident Scenarios for Trend Analysis, Training and Supporting

J. Ortiz-Villafuerte, ININ (Mexico), presented on an initiative to generate a Fast Response Guide to be used in an accident environment based on a collection of analyses derived from the Laguna Verde PSA. These include SBO's, LOCA's and other scenarios. CNSNS has developed some proficiency and experience in the analysis of these scenarios. They have developed a model that can be interactively adjusted to simulate actions and events of an accident such as SAM actions by operators. The guide is a document to support rapid response operations and training.

Multiple representatives indicated they too had tools that were similar to those under development, which meet a similar need. The tools can provide an indication of the state of the plant based on current conditions. Discussions indicated the regulatory attempts to understand the decisions a utility makes during a severe accident but it is the utility that is responsible for those decisions.

I–4.2. Experience and Competence in Severe Accidents Research and Application

This presentation by Yangxiaoming, CNNC (China), describes the accident response and mitigation capabilities. Severe accident simulators are described. One of the simulators is MELCOR based for a three loop PWR. Additionally, a MAAP based simulator is described. Another product is called 'Intelligent SAMG'. It is a software based tool for assisting SAMG implementation. On-line management of severe accident based on the MAAP code. The effectiveness of SAMG is being studied as well.

Questions involved how the code worked, that is understanding why a large break LOCA requires more work but a small break LOCA provides adequate results. It was explained that this is an aspect of the performance of the code used to perform the calculation. The data for the tool is design data and data taken from the plant after it begins operation. Further questions involved trying to understand the effectiveness of SAMGs, how they are implemented and how are SAMGs being performed now. It was explained that the tool is used in the plant for emergency exercises, in the control room for analysis. It is to provide support in specific situations.

I–4.3. The Methodology for Fission Products Release Evaluation for VVER-1000 Under Analytical (Emergency) Center of Russian Regulatory Authority Technical Support

This presentation by G. Arbaev, SECNRS (Russian Federation), describes a methodology for estimating potential source term, potential course of accident and potential consequences of an accident. In this methodology a set of scenarios is run with different assumed time delay between shutdown and severe accident. The ASTEC 2.1 code has been used to produce some of these analyses for a VVER 1000 reactor, modelling horizontal heat exchangers, pressurizers, etc. These are fast-running models to help in estimating time before core degradation begins. After core degradation is judged to have happened, several pre-calculated severe accidents are examined to select the most likely scenario for forecasting purposes.

Discussion revolved around what was meant by faster than real time performance and how the tool is used for severe accident decisions. It was explained that multiple models were run, one for real time, the other faster than real time. Performing faster than real time calculations allows the analyst to perform exploratory calculations for prognosis while the real time calculation is running. At the onset of core damage, the decisions makers pick a predefined calculation scenario that most closely matches the behaviour of the ongoing accident, which is informed by both experts supported by analyses. This is acceptable as the analyst is only checking the potential outcomes to understand the decisions a utility is making.

I–4.4. Multi-Physics Simulator for Core Degradation and Melt Progression in Light Water Reactors During Severe Accidents

A multi-physics code under development at University of Utah were described in this presentation by R. Sisson, SNL (USA). The modern programming practices are highlighted in this university's work illustrating methods and techniques of coupling multi-physics codes such as thermal hydraulics and neutronics. It was suggested that some benchmarking be done to known solutions as a way of evaluating the robustness of this coupling.

Representatives suggested how to potentially present the examples in a better way, namely by reversing the order of the example slides, and questioned if it would be possible to construct models using the input as defined in some of the current severe accident graphical user interfaces. A suggestion was also made to benchmark the results against some of the analyses that may be currently available, or use a smaller problem definition.

I–4.5. Technical Session 3 discussion

The summary discussion of this session is as follows:

(a) The general consensus was that the various representative countries have regulatory bodies which are able to monitor and perform analyses to try to understand the decisions made by utilities during severe accidents, but those bodies cannot tell the utilities what decisions to make. The tools presented assist those bodies in understanding the accident progression during the severe accident. The methods presented involved performing calculations on selected scenarios informed by PSAs. The decisions maker can then attempt to understand the current state by consideration of the library of calculations. The reduction in accuracy of the overall calculation is acceptable as the regulator is trying to understand the decisions of the utility, and they are limited by not being able to tell the utility how to precede during the accident. But it is important to understand the impact of modelling simplifications on the accuracy to ensure the simpler model is still capable of capturing the accident progression.

(b) One method involved the creation of a document which would allow a decision maker, who may not be an analyst, the ability to understand the current state of the reactor.

(c) Additional, the codes allow the user to interactively alter the conditions to vary the boundary conditions during the calculations. This not only allows operators to train on how to handle severe accident scenarios, but it is possible to operators to use these tools to understand the potential impact of decisions in made during the severe accident.

(d) Group discussions focused on the need for potential users of a severe accident simulator to understand the limitations. There is high confidence is the thermal hydraulic calculations used within simulators, but the transition to the simulation of a severe accident introduces a lot of uncertainty, and a potential user must understand that. Although this is more this is more acceptable for the user who is trying to understand multiple scenario decisions. Generally, it was agreed that severe accident simulators would be valuable tools, and severe accident training would also be helpful, but the uncertainty in the codes must be stressed and understood by those using the tool. Namely minor alterations in decisions and timing changes the boundary conditions of the calculation which could potentially have a drastic impact on the results.

I–5. DISCUSSION SESSION SUMMARIES AND RECOMMENDATIONS

Table I–1 outlines the status and needs for severe accident codes use, resulting from Technical Meeting discussions. Additionally, specific recommendations to the IAEA for addressing these needs are included.

TABLE I–1. STATUS, NEEDS AND RECOMMENDATIONS

ACTIVITY	WHAT IS THE STATUS AND WHAT ARE THE NEEDS FOR CODE USERS IN ORDER TO REDUCE THE USER EFFECT
User training	**STATUS:** — Users confirm the importance of code training and the need for a training by code developer's team; — Several user training courses were already conducted, but inside the code user club. **NEED:** — Continue to support the user training inside code user club but extend also in the international framework; — Coupled training with high level training on the physical phenomena is needed. — QUESTION about user certification: few weeks training and with problems to solve. MAAP suggests that utilities include training within their engineering training programmes. — Operating the code is not enough, the interpretation of the results IS important! **Recommendation to IAEA:** **General training on severe accident codes application and uncertainty, in suggesting the best practices approach**: — The need is not for training on how to use the code but on best practices and advanced principles involved regardless of the code itself. What needs to be considered is how the result is interpreted.

TABLE I–1. STATUS, NEEDS AND RECOMMENDATIONS (cont.)

<table>
<tr><th>ACTIVITY</th><th>WHAT IS THE STATUS AND WHAT ARE THE NEEDS FOR CODE USERS IN ORDER TO REDUCE THE USER EFFECT</th></tr>
<tr><td>Participation in technical exchange/user groups</td><td>STATUS:
— Code user clubs are necessary platform to discuss, exchange ideas between code users and code developers. Several code users platform are active.

NEED:
Confirm the importance of the existing code platforms and recommend intercommunications between the platforms;

— International severe accident code user club platform?
— In which user group the user should apply (different type of user: industrial, academic, etc.)?

<u>Recommendations to IAEA:</u>

Since one of the main issues in the use of the codes is the user effect, the following are the recommendations:

A. International Collaborative Standard Problem is suggested:

— New experimental campaign, on a topic of interest, and related code analysis (double blind, blind and open calculation); Possible topics of interests: IVMR, lower head failure (effect of failure and relocation through a failure of solid or mostly solid core debris), core catcher.
— Code to code comparison to assess the code modelling differences and the user effect.

B. CRP Benchmark analysis with severe accident codes on:

— Natural convection/circulation in spent fuel pools under loss of cooling and loss of coolant condition;

— In-vessel hydrogen generation assessment using different codes/ different models and comparison of results for generic PWR with observations and convergence for proper oxidation models;

— Structural integrity assessment of a pressuriser surge line under the SBO;

— Assessment of the late in-vessel phase of severe accident progression.

For a specified severe accident scenario, in one common code benchmarking activity two main technology groups can be defined (depending on the reactor type):

— PWR
— BWR

For both reactor types/groups severe accident simulations for the same severe accident scenario (entire SA case, from initiating event until RPV failure) are to be performed.

Focus is to assess the current status of the severe accident codes predictability in modelling specifically the late in-vessel phase of a severe accident progression.

— Identification of gaps in the code models, code capabilities, phenomena not covered by the SA codes for the late in-vessel phase of severe accident progression. Possible recommendations for severe accidents code improvement.</td></tr>
</table>

TABLE I–1. STATUS, NEEDS AND RECOMMENDATIONS (cont.)

<table>
<tr><th>ACTIVITY</th><th>WHAT IS THE STATUS AND WHAT ARE THE NEEDS FOR CODE USERS IN ORDER TO REDUCE THE USER EFFECT</th></tr>
<tr><td>Review of user guidelines and documentation</td><td>STATUS:
— Users need code user guidelines and user guidelines are already available for each code.

NEED:
— It is recommended to the user to follow the user guidelines;
— It is recommended to the code developer to:
(a) link the user guidelines with the physics of the process;
(b) suggest a nodalization strategy to have a compromise between analysis detail and computational time;
(c) distribute generic full NPP’s input decks.

Unrealistic expectation on what the code can do is a source of confusion in terms of quality of analyses and tools’ application.

<u>Recommendations to IAEA:</u>

Code developers to create a list of best practice cases in the framework of IAEA to be shared with the users (examples cases that the developer provide working together). This activity may result in developing the new IAEA Nuclear Energy Series Report. Best practices approach to include:
— Users define needs in terms of the types of problems that they would like the severe accident codes to model. This includes key phenomenology and accident scenarios (like spent fuel pools, hydrogen production/oxidation, lower head failure, recover during accidents, or after core melt and relocation). The user must define the needs.
— Developers then comment on how ‘best practices approaches’ in using the codes for these cases, as well as concerns about using the models outside of the range of validity and considerations of uncertainty. That way the user understands better what confidence to have in codes’ results.</td></tr>
<tr><td>SA code improvement</td><td>STATUS:
Codes are robust for the range of applicability.

NEED:
Assess numeric point of view (influence of time step on the results, etc.)

<u>References:</u>

IAEA:
— INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, General Safety Requirements No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016):
— INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety Analysis for Nuclear Power Plants, Specific Safety Guide No. SSG-2, IAEA, Vienna (2009).
— INTERNATIONAL ATOMIC ENERGY AGENCY, Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants, Safety Report Series No56, IAEA, Vienna (2008).
— INTERNATIONAL ATOMIC ENERGY AGENCY, Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation, Safety Report Series No.52, IAEA, Vienna (2008).

Russian Federation:
— Общие положения обеспечения безопасности атомных станций, Федеральная служба по экологическому, технологическому и атомному надзору, НП-001-15.
— Требования к составу и содержанию отчета о верификации и обосновании программных средств, применяемых для обоснования безопасности объектов использования атомной энергии, Федеральный надзор России

Finland:
— DETERMINISTIC SAFETY ANALYSES FOR A NUCLEAR POWER PLANT, STUK, Guide YVL B.3 / 15 November 2013.</td></tr>
</table>

TABLE I–1. STATUS, NEEDS AND RECOMMENDATIONS (cont.)

ACTIVITY	WHAT IS THE STATUS AND WHAT ARE THE NEEDS FOR CODE USERS IN ORDER TO REDUCE THE USER EFFECT
Graphical User Interface (develop input deck and post processing of the data)	**STATUS:** — In general, the users with long experience (several years of experience, e.g. more than 10 years of experience) with the code use native format for developing the input deck (e.g. ASCII), and prefer to use native format. In general, new users use GUI from the beginning. — Different GUI tools are available; some are already in a mature state, some need improvements. **NEED:** — Continue to support the development of GUI in order to have a more user friendly code; — GUI should support the nodalization development and post processing analyses. **Recommendations to IAEA:** **No recommendations**.
Uncertainty analyses (uncertainty methodology and tools)	**STATUS:** — Some uncertainty analyses are in progress and relevant examples are available in the public international scientific technical literature; — It should be a common practice in research framework. **NEED:** — Elaboration of common practices for performing sensitivity and uncertainty analyses; — Automatic coupling between uncertainty tools and codes; — Which is the recent approach that we should follow when we do severe accident analyses (sensitivity and uncertainty); — Express needs for utilities to do **uncertainty analysis** and do it properly and affordably. **Recommendations to IAEA:** **Plan a next meeting about the use of uncertainty in severe accident analysis — not how to do UA, but how uncertainty is handled in severe accident analysis and severe accident response training** (this could be coupled with other consistent initiative in other framework) **Develop a new CRP and benchmarking of the codes in this area.**

Annex II

CONTENTS OF THE INCLUDED CD-ROM

Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors IAEA Headquarters Vienna, Austria, 9–12 October 2017

No.	TITLE
0	Papers Submitted to the Technical Meeting on the Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors

TECHNICAL SESSION I: COMPUTER CODES AND MODELS FOR EVALUATION OF SEVERE ACCIDENTS IN WATER COOLED REACTORS

No.	COUNTRY	TITLE	PRIMARY AUTHOR	TECDOC SECTION
1	Germany	Status of Development of GRS Code System AC^2: Part I: Modelling of Reactor Phenomena	L. Lovasz	2.1.1 2.2.1 3.1
2	Germany	Status of Development of GRS Code AC^2: Part II: Modelling of Containment Phenomena	C. Spengler	2.1.2 2.2.1 3.1
3	United States of America	Using Deterministic and Probabilistic Methods for MELCOR Severe Accident Uncertainty Analysis	P.D. Mattie	2.2.5 3.1
4	India	Severe Accident Modelling and Analysis of VVER 1000 (V-412) for Kudankulam NPP, India	K. Abhishek	3.1
5	United States of America	Status of MELCOR 2.2 and Plans for MELCOR 3.0	R. Gauntt	2.2.5 2.3 3.1
6	United States of America	ASTEC Model Development for the Severe Accident Progression in a Generic AP1000	L. Albright	2.1.1 2.2.2 3.1
7	Germany	Core Degradation Analysis for a Generic German PWR with the Severe Accident Code ATHLET-CD	P. Wilhelm	2.2.1 3.1
8	Italy	ASTEC, MAAP and MELCOR Benchmark Code Analysis of an Unmitigated SBO Transient in a PWR-900 Like Reactor	F. Mascari	2.2.2 2.2.3 2.2.5 3.1
9	Lithuania	Lithuanian Energy Institute Experience in Simulation of Severe Accidents in Water Cooled Reactors	T. Kaliatka	3.1
10	Russian Federation	Status of Development and Applications of SOCRAT Code for Severe Accident Simulation	A. Kiselev	2.2.7 3.1
11	United States of America	Findings from Uncertainty Studies Evaluating Severe Accident Phenomena and Off-site Consequences Using MELCOR and MACCS	A. Hathaway	2.2.4 2.2.5 3.1
12	Belgium	Ex-vessel Combustible Gas Generation	A. Bieliauskas	2.1.2 3.1
13	Ukraine	Application of Severe Accident Analyses Codes in Safety Justifications and Regulatory Review for Ukrainian NPPs	D. Gumenyuk	3.1
14	Hungary	Using New Versions of Severe Accident Codes for VVER-440/213 Type Nuclear Power Plants	A. Nemes	3.1
15	France	Overview of the Integral Code ASTEC V2.1 Revision 1	A. Bentaib	2.2.2 3.1

No.	COUNTRY	TITLE	PRIMARY AUTHOR	TECDOC SECTION
16	United States of America	MELCOR 2.2 Analyses of the Accidents at Fukushima with Emphasis on Source Term and Key Lessons Learned	R. Gauntt	2.2.5 3.2
17	Spain	On the Applicability of Severe Accident Codes as Forensic Tools: A Study on the Unit 1 of the Fukushima Site	C. Lopez	3.2
18	Jordan	Investigation of Performance of Severe Accident Safety Features for Advanced Reactors to Cope with Fukushima Accident and Post Fukushima Requirements	S. Melhem	3.2
19	Mexico	BWR Mark II LOCA DBA Severe Accident Simulation with RELAP/SCDAPSIM	J. Ortiz-Villafuerte	2.2.6 3.2
20	France	The Fukushima-Daiichi Accident Computations with ASTEC and How to Match it with the Dose Measurements in the Environment	A. Bentaib	2.1.2 2.2.2 3.2
TECHNICAL SESSION III: BASIC PRINCIPLE SIMULATORS FOR SEVERE ACCIDENTS				
No.	**COUNTRY**	**TITLE**	**PRIMARY AUTHOR**	**TECDOC SECTION**
21	Mexico	Development of a Fast-Response Guide for Specific Simulations of Severe Accident Scenarios for Trend Analysis, Training and Supporting	S. Mugica	3.3
22	China	Experience and Competence in Severe Accident Research and Application	Yangxiaoming	3.3
23	Russian Federation	The Methodology for Fission Products Release Evaluation for VVER-1000 Under Analytical (Emergency) Center of Russian Regulatory Authority Technical Support	G. Arbaev	3.3
24	United States of America	Multi-Physics Simulator for Core Degradation and Melt Progression in Light Water Reactors During Severe Accidents	R. Sisson	3.3
25	Brazil	Simulation of a Station Blackout at the Angra 2 NPP With MELCOR Code	L. Nelbia	3.3
26	IAEA	SAMG-D: The IAEA Training Toolkit on the Development of Severe Accident Management Guidelines	I. Khamis	3.3

Annex III
OVERVIEW OF RELATED IAEA PUBLICATIONS

III–1. INTRODUCTION

The IAEA has created several publications which describe safety analysis methodology. These publications provide guidance for the performance and applications of accident analyses, with some directly to severe accidents. Many of these publications build on those published previously in order to add supplementary information specific to different reactor types or additional applications. Of potential interest for readers of this publication, relating directly to safety and accident analysis, the following IAEA publications are summarized.

III–2. SAFETY REPORTS SERIES No. 23: ACCIDENT ANALYSIS FOR NUCLEAR POWER PLANTS

This report was developed as a result of several consultancy meetings, was reviewed at the IAEA Technical Committee Meeting on Accident Analysis in Vienna, 30 August–3 September of 1999, and was published in 2002.

This report includes guidance on the performance of reactor accident analyses, including selection of events and criteria, computer codes and models, preparation of data, analysis of calculated results, and suggestions for improving the quality of an analysis. It describes the different types of codes for accident analysis, including both those for design basis and beyond design basis accidents, as well as explains many important considerations for their use. Within Annex IV of this Safety Report are several examples for each type of accident code described, including:

— Reactor physics;
— Fuel behaviour;
— System thermohydraulics;
— Containment;
— Structural analysis;
— Mechanistic system thermohydraulics;
— Parametric codes.

This publication is intended to apply to operating and under construction nuclear power plants and deals with internal events and associated systems in both design basis and beyond design basis scenarios. Discussed are both conservative and best estimate approaches to accident analysis. Neutronic, structural, as well as radiological aspects are discussed, though the focus of the publication is on the thermohydraulic.

III–3. SAFETY REPORTS SERIES No. 56: APPROACHES AND TOOLS FOR SEVERE ACCIDENT ANALYSIS FOR NUCLEAR POWER PLANTS

This publication was written in 2008 as complementary report to *Safety Reports Series No. 23*: *Accident Analysis for Nuclear Power Plants*, written specifically to further detail the phenomena and considerations for severe accident analysis. Though the previous publication includes coverage of design basis accident analysis and beyond design basis accident analysis, the guidance specifically provided for severe accident analysis is limited.

This report lists important components of severe accident analysis, provides an overview of the modelling of phenomena involved in severe accidents, including both in-vessel and ex-vessel phenomena, and explains the different approaches to severe accident analysis taken by each type of code. Additional uses of severe accident analyses are discussed, including use for:

— Training purposes;
— Development/validation of accident management programmes;
— Design/validation of severe accident mitigation systems;
— Reactor plant simulators.

Within the appendices of this publication is a demonstrational example of the steps, as recommended in the main text, for severe accident analysis. In the included annexes, the features of select severe accident codes are discussed (ASTEC, ATHLET-CD, ICARE/CATHARE V1, MAAP 4.03, MELCOR 1.8.4, and SCDAP/RELAP5/MOD3.2).

III–4. SAFETY STANDARDS No. SSG-2: DETERMINISTIC SAFETY ANALYSIS FOR NUCLEAR POWER PLANTS

This guide, published in 2009, discusses the IAEA recommendations and guidance for the use and application of deterministic safety analysis in areas including:

— Design;
— Licensing;
— Assessment of safety analysis reports;
— Plant modifications;
— Analysis of operational events;
— Development and validation of emergency operating procedures;
— Development of severe accident management guidelines;
— Periodic safety reviews.

The publication additionally provides high level explanations of event categories and plant states and of acceptance criteria. It describes two differing approaches to deterministic safety analysis: conservative deterministic and best estimate plus uncertainty. Also described are the recommended steps for verification and validation processes for computer codes used for safety analysis and source term evaluation for operational states and accident scenarios.

A revised publication is expected to be published in 2019.

III–5. SAFETY STANDARDS No. NS-G-2.15: SEVERE ACCIDENT MANAGEMENT PROGRAMMES FOR NUCLEAR POWER PLANTS

This guide, published in 2009, outlines the IAEA recommendations and guidance for accident management programmes, including in management of severe accidents.

The publication establishes the concept and criteria of accident management programmes at a high level. This includes setting requirements for an acceptable accident management programme, definition of accident management stages and objectives,

This publication further provides information on accident management programme development, including development of management strategies and development of procedures and guidelines. Among the contents are descriptions of recommended steps in severe accident analysis, using accident analysis computer codes, to support the creation of procedures and guidelines.

A revised publication is expected to be published in 2019.

III–6. IAEA-TECDOC-1351: INCORPORATION OF ADVANCED ACCIDENT ANALYSIS METHODOLOGY INTO SAFETY ANALYSIS REPORTS

This publication was written in 2003 to complement the *Safety Report Series No. 23, Accident Analysis for Nuclear Power Plants*, by providing further guidance in the application of computer codes for accident analysis used to develop Safety Analysis Reports (SARs). This publication is intended for use by developers and reviewers (utility and regulator) of the deterministic safety analysis in the SAR of a nuclear power plant.

Included in this publication is an overview of a variety of advanced computer code categories including:

- Thermal hydraulic system;
- Reactor dynamics (including coupled);
- Containment thermal hydraulics;
- Severe accident analysis;
- Others (fuel behaviour, CFD, fire analysis, etc.).

Beyond the code description, this publication covers:

- Accident analysis development for use in the accident analysis chapter of a SAR;
- Basic design analysis through support of system design, structural analysis, radiation protection, and fuel design and management in the SAR;
- Management of uncertainties through validation of models, quantization of existing uncertainty, and minimization of user uncertainty.

The appendices of this publication further provide a guide for preparation of a safety analysis, recommendations for changes of nodalization, an overview of data transfer and management of interface for pressurized thermal shock analysis, a description of the assumptions in licensing analyses, and an example of the standard SAR format and content.

III–7. IAEA-TECDOC-1594: ANALYSIS OF SEVERE ACCIDENTS IN PRESSURIZED HEAVY WATER REACTORS

This publication was written in 2008 to complement *Safety Reports Series No. 56, Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants*. Several factors important to severe accident analysis are described, an overview of phenomena and modelling is given, available computer codes are categorized and differences in approach to severe accident analysis in pressurized heavy water reactors (PHWRs) described to supplement that of the Safety Series publication. The Safety Series focuses primarily on pressurized water reactor and boiling water reactor designs, which can only serve as preliminary guidance for PHWRs, and this TECDOC addresses this with further guidance specific to PHWR designs.

This publication includes descriptions of Canadian and Indian PHWR designs, severe accident related phenomena of concern to PHWR systems, the steps for analysis of severe accidents in PHWRs, failure criteria, and PHWR specific accident management.

Within the appendices of this publication are descriptions of the main PHWR severe accident experimental facilities, main features of severe accident codes (MAPP4-CANDU, ISAAC), and severe accident code use in India. Further, the annexes of this publication include example results of severe accident analysis and provides an overview of the computer codes used for modelling accidents with limited core damage, including the areas:

- Reactor physics (WIMS-AECL, RFSP);
- Thermal hydraulic analysis (CATHENA);
- Fuel analysis (ORIGEN, ELESTRES, ELOCA);
- Fission product analysis (SOURCE, SOPHAEROS);
- Moderator analysis (MODTURC_CLAS);
- In-core damage analysis (TUBRUPT);
- Containment analysis (GOTHIC, SMART);
- Dose assessment (ADDAM).

III–8. IAEA-TECDOC-1727: BENCHMARKING SEVERE ACCIDENT COMPUTER CODES FOR HEAVY WATER REACTOR APPLICATIONS

This publication, published in 2013, summarizes the results of the coordinated research project (CRP) on Benchmarking Severe Accident Computer Codes for HWR Applications.

The benchmark scenario used for this CRP consists of a reference generic CANDU-6 power plant subject to a station blackout. Following the benchmark description, failure criteria are defined for a number of component failure mechanisms. Participating institutions utilized a variety of codes in simulating the benchmark scenario, and results were compared. The benchmark served as a basis for gaining understanding the code limitations and uncertainties as well as for observing user effects between some institutions which performed the benchmark with common code suites. The following is a list of participating institutions and corresponding code(s) used:

- Atomic Energy of Canada Limited (AECL)
  MAAP4-CANDU v4.0.6A;
- Bhabha Atomic Research Center, Reactor Engineering Division (BARC-RED)
  RELAP5 Mod 3.2, ANSWER, CAST3M, MELCOOL;
- Bhabha Atomic Research Center, Reactor Safety Division (BARC-RSD)
  SCDAP/RELAP5 Mod 3.2, PHTACT, ASTEC;
- Korea Atomic Energy Research Institute (KAERI)
  ISAAC 4.02;
- Nuclear Power Corporation of India Limited (NPCIL)
  ATMIKA-T, CONTACT, SEVAX, MCCI, PACSR/STAR, ACTREL;
- Politechnical University of Bucharest (PUB)
  SCDAPSIM/RELAP5 Mod 3.4;
- Shanghai Jiao Tong University (SJTU)
  SCDAP/RELAP5 Mod 3.4.

Annex IV

IAEA PROJECTS AND ACTIVITIES ON SEVERE ACCIDENT CODES AND ANALYSES

IV–1. INTRODUCTION

The IAEA project on severe accidents codes and analyses for WCRs facilitates advancement of the state of practice in code applications in Member States with the goals to:

- Improve phenomenological understanding of severe accidents and the capability to analyse them;
- Provide better understanding and characterization of sources of uncertainty and their effect on the key figure of merit prediction uncertainty from severe accident codes;
- Promote information exchange on the advancements in simulation models and codes for severe accidents;
- Educate newcomers interested in simulation and modelling of severe accidents.

IV–2. COORDINATED RESEARCH PROJECTS

The IAEA Coordinated Research Projects bring together research institutes in both developing and developed Member States to collaborate on research topics of common interest with the goal to contribute towards greater understanding or resolution of a specific issue or problem. In the broadest sense, the CRPs cover transfer of knowledge and tools and their applications based on nuclear and related technologies.

IAEA-TECDOC-1727: Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications, issued by the IAEA in 2013, summarizes the results from the CRP on benchmarking of severe accident analysis codes used for the analysis of severe core damage accidents in HWRs. The exercise promoted international collaboration among IAEA Member States to improve the phenomenological understanding and analysis capability of severe core damage accidents. The scope included the identification and selection of a severe accident sequence, selection of appropriate geometrical and boundary conditions, conducting benchmark analyses, and comparing the results of all code outputs, evaluating the capabilities of existing computer codes to predict important severe accident phenomena and suggesting necessary code improvements and/or new experiments to reduce uncertainties. The objective of the CRP was to conduct a benchmark exercise on severe accident computer codes used for consequence analysis of HWRs. The purpose was to compare the integrated effects of embedded models in the codes, gain an understanding of their limitations, assess the level of uncertainties, and thereby increase the confidence in severe accident code predictions. The code to code comparisons provided the justification required for model improvement and reduction of uncertainties.

The CRP on Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for the Severe Accident Analyses in Water Cooled Reactors to be launched in 2019 and completed in 2024 focuses at advancing the understanding and characterization of sources of uncertainty in severe accident codes for WCRs. Sources of uncertainty include (1) epistemic uncertainties from lack of knowledge, reflected in imperfect models in the codes — ‘model–form uncertainty’, (2) aleatory/stochastic/random uncertainties in boundary and initial conditions, and (3) so called cliff–edge effects that result in bifurcation of the accident progression. Effects of the first two sources of uncertainty can be assessed with best-estimate plus uncertainty methods, while the third may require probabilistic methods. A major outcome of this CRP will be to raise the level of expertise and sophistication of severe accident code users and support the proper interpretation of code results, including their uncertainty. This CRP will further the current state of knowledge on severe accident modelling and will address important predictive uncertainties relevant to severe accident progression and mitigation in deployed and advanced WCRs. Coordination by the Agency will provide significant added value through objective and peer reviewed evaluations, by means of benchmark studies by the participating Member States, and thus will lead to new knowledge and sharing of research results relevant to application of severe accident codes. The newly developed knowledge resulting from the research activities supported by this CRP will be transferred to developing countries through specific training workshops and educational courses.

IV–3. TECHNICAL MEETINGS

In addition to the meetings related to post Fukushima accident reassessment of severe accidents propagation and consequences and the follow up meetings on severe accidents and modelling as described in Section 1.1., two other relevant Technical Meetings were held.

The Technical Meeting on Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling was held in Shanghai, China, 17–21 October 2016. This meeting allowed Member States to share information on recent R&D activities related to in-vessel melt retention and ex-vessel corium cooling for severe accidents in water cooled reactors. 52 nominated participants from 18 Member States attended the meeting and 11 observers from the host country participated. In total, 33 presentations were given, discussing the following topics:

- General considerations on in-vessel melt retention strategy;
- External reactor vessel cooling;
- Molten pool behaviours and structural integrity of reactor vessel;
- Application of in-vessel melt retention to specific reactor designs;
- General consideration on ex-vessel corium cooling strategy;
- Application of ex-vessel corium cooling to specific reactor designs;

The meeting served to highlight the developments in in-vessel melt retention and ex-vessel corium cooling strategies, particularly through understanding of key phenomena, experimental and analytical studies, improvement and validation of codes and application to specific reactor designs. Meeting participants pointed out that international collaboration is necessary for the development of a common understanding of relevant phenomenology and technology.

The Workshop on Advances in Understanding the Progression of Severe Accidents in Boiling Water Reactors was held in Vienna, Austria, 17–21 July 2017. The purpose of this workshop was to provide a platform for experts from Member States and international organizations to exchange and disseminate information on R&D activities regarding the progression of severe accidents in BWRs, including updated information on the Fukushima Daiichi NPP accident. The workshop aimed to facilitate the exchange of relevant R&D results, foster worldwide collaboration in R&D activities, enhance communication between industry (utilities, vendors, etc.), regulatory bodies and research organizations, and discuss and update scientific and engineering knowledge in this area. The workshop was attended by 28 nominated participants from 13 Member States, consisting of experts in severe accidents, a representative from the OECD/NEA, and several IAEA staff members. In total, 34 participants presented and discussed the following topics:

- Forensic investigation and analyses of the Fukushima Daiichi NPP accident, addressing such aspects as core damage progression, corium/debris relocation, degraded core cooling by external water injection, and containment cooling and venting;
- Experiments to better understand key phenomena during severe accidents (e.g. core melt and debris formation, corium cooling, molten core–concrete interaction, hydrogen generation and transport, and pool scrubbing);
- Development/improvement of severe accident analysis models and codes (e.g. core melt, reflooding of degraded cores, in-vessel melt retention, ex-vessel corium cooling, hydrogen combustion, and containment response);
- Benchmarking of analysis models and codes (benchmarking with experimental data and/or code to code benchmarking), and verification and validation of analysis codes;
- Improved severe accident scenarios for BWRs, and the relevant strategies and technologies to prevent the progression of an accident and mitigate the consequences.

The workshop pointed that globally, majority of R&D activities related to severe accidents had been focused on PWRs mainly due to the dominant number of operating reactors until the Fukushima Daiichi NPP accident in March 2011. The accident has activated/revitalized R&D efforts on BWR designs in Member States and international organizations, achieving a significant progress in understanding the severe accidents' phenomenology in BWRs. However, there are still several remaining areas to be addressed for further improvement in understanding progression of severe accidents in the BWR plants.

IV–4. TRAINING COURSES

The IAEA offers a large number of training courses to assist in human capacity building among Member States. These courses have a wide range of topics including nuclear engineering safety systems, human resource development, and severe accidents progression and management guidelines development among others. Manly these courses include lectures with extensive practical learning using basic principle NPP simulators and the severe accident management guideline development (SAMG-D) toolkit.

The IAEA Nuclear Power Technology Development Section (NPTDS) maintains a suite of educational and training basic principle simulators and has provided several training courses to Member States on their use. Basic principle simulators are designed to demonstrate nuclear engineering concepts and plant behaviour, however with less rigorous and computationally intensive models than those used in plant safety analysis or in full scope plant simulator. Many have also incorporated accident scenarios and this functionality allows the simulators to be used as educational tools for understanding accident progression and plant response. *IAEA-TECDOC-1836: Developing a Systematic Education and Training Approach Using Personal Computer Based Simulators for Nuclear Power Programmes* lists training courses the IAEA held using basic principle simulators between the years 1999 and 2017, and among these are training courses directly focused on reactor technologies and severe accidents. Lessons in these training courses include topics such as the basic phenomena of design basis and severe accidents, the plant response to transients, and the stages of accident progression. Lesson concepts are reinforced by participant direct use of basic principle simulators in solving both normal operating and accident transient states of a particular NPP type, such as PWR, BWR, VVER, PHWR or iPWR.

The IAEA has held several workshops on the SAMG-D toolkit. Nuclear power plants are designed to withstand many types of incident events while maintaining safe operation through use of plant safety systems and emergency operating procedures (EOPs). However, in the case of a severe accident, where the plant is beyond a recoverable state and fuel damage is expected, steps can be made to mitigate the effects of the accident. SAMGs are used to provide systematic guidance to operators toward mitigation. The IAEA SAMG-D toolkit provides information to assist in the development of SAMGs by providing educational information on nuclear safety, accident management, severe accident phenomena, and accident mitigation. It also describes strategies for the development and implementation of SAMGs including suggested plant analysis, methods for determining accident prioritization, suggestions for effective transition from EOPs, and recommends nuclear plants to form of a Technical Support Centre and a severe accident training program. SAMG-D toolkit workshops instruct participants on the effective navigation and use of the toolkit in addition to providing an inclusive description of accident management methodology.

The IAEA has held several joint training courses in coordination with the Abdus Salam International Centre for Theoretical Physics (ICTP) on diverse scientific topics of mutual interest such as advanced reactor technologies, passive safety systems and the use of basic principle simulators in teaching on advanced water cooled reactor designs. These training courses include many which are comprehensive to reactor design and technology, where some individual lessons are dedicated to severe accidents. Of notable relevance are lessons which cover reactor safety analysis, severe accident phenomena, and reactor modelling and simulation. Some of these joint training courses are focused in scope to single topics — for example, a training course dedicated entirely to the phenomena (physical, chemical, and radiological) which occur in the progression of a severe accident.

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