

PROJECT FINAL REPORT

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5	Budapest University of Technology and Economics	BME	HU
6	Commissariat à l'Energie Atomique et aux Energies Alternatives	CEA	FR
7	Ricerca sul Sistema Energetico SpA	RSE	IT
8	Chalmers tekniska högskola AB	CHALMERS	SE
9	Centro de Investigaciones Energeticas Medio Ambientales y Tecnologicas	CIEMAT	ES
10	National Centre for Scientific Research "DEMOKRITOS"	DEMOKRITOS	EL
11	Electricité de France SA	EDF	FR

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45	Department of Atomic Energy	BARC	IN

1. Final publishable summary report

1.1 Executive summary

After four years of operation in the 6th Framework Programme (FP6) of Research and Development of the European Commission, the SARNET (Severe Accident Research NETwork of excellence) network has continued its activity between 2009 and 2013 with financial support of FP7 as the SARNET2 project. Forty-seven organisations (research, universities, industry, utilities, safety authorities and technical safety organizations) from 24 countries have networked their capacities to resolve the most important remaining uncertainties and safety issues on severe accidents in existing and future water-cooled nuclear power plants. The network, coordinated by IRSN (France), tackles the fragmentation that may still exist between the different national R&D programmes by defining common research programmes and developing common tools and methodologies for safety assessment. The overall work, involving about 250 researchers and 30 PhD students, represents an equivalence of 40 full-time persons per year. Most key European R&D actors are members and the network takes also benefit from the knowledge and work of important non-European organizations from USA, Canada, Korea, India and Japan.

The research priorities were periodically updated by an experts' group that accounts for the results of recent research and, since 2011, for the impact of Fukushima accidents.

The collaborative work on corium, containment and source term phenomena has allowed a significant progress on the following 6 high-priority issues: corium/debris coolability, molten-core-concrete-interaction, steam explosion, hydrogen combustion in containment, impact of oxidising conditions on source term, and iodine chemistry. New experiments have been funded and performed on debris bed reflooding, molten-core-concrete-interactions and source term. Knowledge on severe accident phenomena and management was continuously produced through joint interpretation of past and new experiments, benchmark exercises between codes, state-of-the-art reports and elaboration of physical models.

This knowledge was capitalised in common tools: ASTEC IRSN-GRS integral code for simulation of severe accidents through new physical models, and DATANET database through storage of experimental data and reports using the JRC STRESA tool (today 265 experiments in 43 facilities).

Dissemination of knowledge was also an important part of the activities during the four years: three education courses for young researchers or students or for staff managers, edition of a textbook of 750 pages on severe accident phenomenology, mobility of researchers or students between the network partners, two ERMSAR periodic conferences (plus the next one in October 2013 under preparation) that are becoming the major worldwide conference on severe accident research, and more than 360 technical papers in conferences and scientific journals.

The SARNET outcomes are very positive: success of networking of R&D activities of diverse types and between diverse types of organizations, consolidation of ASTEC code as European reference, update of R&D priorities accounting for recent international R&D, many deliverables, including state-of-the-art reports, dissemination of knowledge to young generation and new nuclear countries. Collaboration was set up with OECD/CSNI through joint state-of-the-art reports and benchmark exercises between codes.

But the momentum must not be lost, especially after the Fukushima accidents and the need to further consolidate nuclear power plants with new SA mitigation systems. After the end in March 2013 of the SARNET2 FP7 project, self-sustainability of the network is achieved through integration in the NUGENIA European association addressing Gen.II-III R&D, as one of the 8 technical areas. Networking activities will continue in this new framework in the same way as up to now.

1.2 Summary description of project context and objectives

Most of the actors involved in severe accident (SA) research in Europe, plus Canada, Korea, United States, India and Japan (47 partners), have networked in SARNET (Severe Accident Research NETwork of Excellence), through the SARNET2 FP7 project since April 2009, their capacities of research in order to resolve important pending issues on postulated SAs of existing and future Nuclear Power Plants (NPPs). Several organizations have joined the network during the course of the project: KINS, the Korean Technical Safety Organisation at the end of the 1st period, and later on, BARC from India, JAEA and JNES from Japan, NCBJ from Poland and ICL from UK. The project has been defined to optimise the use of the available means and to constitute a sustainable consortium for the development of common research programmes and of a common computer tool to predict the NPP behaviour during a postulated SA (ASTEC integral code, jointly developed by IRSN and GRS). With this aim, the SARNET partners contributed to a Joint Programme of Activities (JPA), which consists of:

- Maintaining and improving an advanced communication tool within the partners,
- Harmonizing and re-orienting the research programmes, and defining new ones;
- Performing experimental programmes on high priority issues, defined during the SARNET FP6 project;
- Analysing experimental results in order to elaborate a common understanding of relevant phenomena;
- Developing the ASTEC code (including its applicability to all types of European NPPs), which capitalizes the knowledge produced within the network in terms of physical models;
- Developing the DATANET experimental database, in which all the results of research programmes are stored in a common format;
- Developing education courses on SAs for students and researchers, and training courses for specialists;
- Promoting personnel mobility amongst various European organizations;
- Organizing periodically a large international conference on SA research (ERMSAR) in order to better share at the international level the knowledge gained through the network.

Six high-priority issues were defined for further R&D:

- Core coolability during reflooding and debris cooling;
- Ex-vessel melt pool configuration during Molten-Corium-Concrete-Interaction (MCCI), ex-vessel corium coolability by top flooding;
- Melt relocation into water, ex-vessel Fuel Coolant Interaction (FCI);
- Hydrogen mixing and combustion in containment;
- Oxidising impact on source term (Ruthenium oxidising conditions/air ingress for high burn-up and Mixed OXide (MOX) fuel elements);
- Iodine chemistry in Reactor Coolant System (RCS) and in containment.

The interpretation circles, gathering most high-level specialists in these domains, have been closely associated to the existing experimental programmes, in particular Phébus FP, International Source Term Programme (ISTP), International Scientific and Technical Centre (ISTC) projects and OECD/NEA/CSNI projects.

Achieved results at the end of the project

After 4 years, SARNET2 has consolidated the sustainable integration of the European SA research capacities, already in progress in the FP6 project. It includes a large majority of the European actors involved in SA research plus a few non-European important ones (some of the latter have joined the network during the project). A few organizations are covering a wide range of competences, though not complete, whereas others are specialized in specific areas: this leads to develop complementarities. The network has continued to reduce the fragmentation that still exists among the different R&D national programmes, notably in defining common research programmes and developing common computer tools and methodologies for NPP safety assessment.

Through a periodic review of priorities and co-programming of work amongst organisations, the use of available means and budget was more efficient. The latest update of priorities accounted in particular for the Fukushima accidents in Japan and allowed harmonizing and re-orienting existing programmes and jointly defining new ones for the near future.

Capitalizing the acquired knowledge in the ASTEC integral computer code (jointly developed by IRSN and GRS to predict the NPP behaviour during a postulated SA) and in the experimental database produced necessary conditions for preserving the knowledge acquired in this field by thousands of person-years and disseminating it to a large number of end-users. By fostering collaborative work on developing and validating ASTEC, the role of Europe as world leader in this domain has been consolidated since there is today an alternative to US technology. The validation level has greatly increased thanks to the activities of the 32 involved partners and most of the models have been improved by IRSN and GRS in the successive versions. The latest one, V2.0rev3, has integrated most of the knowledge produced in the network, and progress was made on the extension of applicability to BWR and CANDU NPP types despite still present deficiencies concerning core degradation modelling (these models are currently being implemented by IRSN and GRS in the next major version V2.1 that is foreseen end of 2014). ASTEC is available for all the SARNET2 members or other organizations that would like to use it for NPP safety assessment or improvement.

The critical mass of competences was achieved to perform small and large-scale experiments, jointly analyse their results in order to elaborate a common understanding of the concerned physical phenomena, develop models and integrate them into SA codes, in priority ASTEC.

Through an education and training programme addressing mainly young scientists, the European excellence in the SA domain has been consolidated on the long term. Synergies were developed with educational institutions, either universities that hosted the educational courses and seconded students in other laboratories, or international entities such as the ENEN network in Europe. This allowed keeping attractive the concerned domain of activity for students and young researchers. The network provides also a wide panel of competences for supporting the emergence of new nuclear countries.

The network activities and results have been presented in many nuclear events worldwide, such as international periodic conferences like ICONE or ICAPP or Forums like EUROSAFE, as well as symposiums organized in Asia (NUSSA in China, SAAM in India). The European Review Meeting on Severe Accident Research (ERMSAR) was organised every 18 months and has become the major worldwide conference on SA research, gathering more than 140 participants in its two last occurrences. A public web site presents up-to-date information to all people interested in the SA subject.

As the end-products developed by the network such as ASTEC and the experimental database may be used not only for R&D activities but also for industrial applications, many European industry and safety authorities (or technical safety organizations) are contributing to SARNET2. In return, the end-products that capitalize the large amount of knowledge acquired in this area contribute to a better

prevention and mitigation of SA in existing and future European NPPs of diverse types, and thus to the improvement of their safety.

The ultimate aim to ensure the long term self-sustainability of the network has been reached by the integration into the NUGENIA international association that is a legal entity with a strong coordinating structure for work orientation in a “virtual centre of excellence”. SARNET clearly has become an international reference for SA research priorities and impacts on national programmes and fund allocations. Progressively all the research activities in this field will become strongly coordinated by the network, which will contribute to an optimised use of European resources.

1.3 Description of the main S&T results/foregrounds

1.3.1 WP4: ASTEC

The WP4 is composed of three sub-WPs: USTI on users' support, training and integration of models, ACAS on code assessment and AMEX on code extension to BWR and CANDU reactors.

WP4.1: Users' Support, Training and Integration (USTI)

The USTI activity aimed at providing a support to ASTEC code users through elaboration, improvement, delivery and maintenance of code versions and revisions, assistance in code use, organisation of training courses ... as well as at enhancing code use experience through organisation of users' club meetings. Its other goal is to allow capitalising the R&D knowledge in the field of SAs by integration of models proposed in WP5 to 8.

IRSN and GRS have successively released to 27 partners the versions of the new V2 ASTEC series (Figure 1): V2.0 rev0 in July 09, V2.0rev1 in June 2010, V2.0rev1p2 in March 2011, V2.0-rev2 in December 2011, and finally a special patch version V2.0-rev2p2 in July 2012, thus becoming the reference version for the last part of the project. A 3rd revision V2.0-rev3 was released in July 2013, thus becoming the new reference version to be used after SARNET2's end. IRSN and GRS performed the regular corrective maintenance of all these versions.

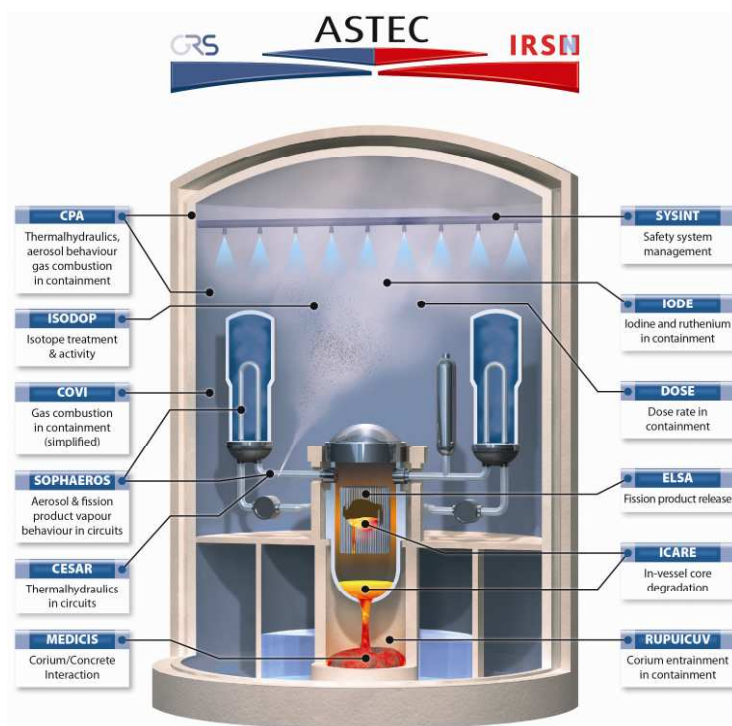


Figure 1: ASTEC integral code structure

The main improvements of this V2 series with respect to the V1 series that was assessed by partners in SARNET FP6 were mainly the applicability to EPR and the advanced models for core degradation issued from the IRSN mechanistic code ICARE2, in particular the 2D magma relocation models. The successive revisions have also capitalised most of the R&D knowledge by integration of models proposed in WP5 to 8. The most important modelling improvements are summarized below: thermal-hydraulic behaviour in swollen water level volumes and condensation processes in the primary/secondary circuits; thermal-hydraulic behaviour at low pressure and low Peclet number (typical of In-Vessel Retention (IVR) and External Reactor Vessel Cooling (ERV)); in-vessel magma oxidation, corium jet fragmentation, heat transfers between corium/debris and vessel lower

head, phase separation model in corium in lower head; behaviour of hydrogen flame front in multi-compartment containment; iodine reactions in containment (in particular Ag/I in the sump, radiolytic oxidation of I⁻ into I₂, interactions with paints...); ruthenium behaviour, behaviour of EPR core catcher. New models have also been implemented: successive corium slumps from the vessel towards the cavity after vessel failure; generic DCH correlations for all French PWRs; model of dry aerosol resuspension in containment; most recent AREVA correlations for passive autocatalytic recombiners and new detailed diffusion-type recombiner model; RCS gas chemistry kinetics; improved heat transfers for dry MCCI, new upper crust model for MCCI and set-up of an adequate thermochemistry data file automatically selecting the most appropriated data with respect to the simulated corium and concrete.

For support to code users, two one-week training courses were organized by IRSN for beginners in Aix-en-Provence in June 2009 and in April 2011. Two ASTEC Users' Club meetings were respectively hosted by GRS in October 2010 in Cologne and by IRSN in January 2013 at Aix-en-Provence in order to enhance code users' experience and promote fruitful discussions with code developers: they gathered about 50 users each time. An ASTEC web site has been created in October 2009 for a direct access to the code versions and documentation, and then was continuously improved.

In parallel, IRSN and GRS have continued working on the elaboration of the second ASTEC V2 major version (version V2.1). The three main structural developments have been completed at IRSN: new coupling of CESAR thermal-hydraulics module and ICARE core degradation module, account for canisters and more generally for in-core multi-coolant flows, and generalisation to all ASTEC modules of the material data bank use. Intensive testing of the future V2.1 version under development with these new features is underway. Besides, the development of a new direct containment heating model is progressing well at GRS. The elaboration of the future V2.1 version will continue after SARNET2's end for a delivery planned at the end of 2014. This version will in particular be fully applicable to sequences in BWR and CANDU NPPs.

With respect to the contract deadlines, there was some delay but not so large than it may appear because IRSN and GRS decided to decrease the frequency of release of major versions in order to assure sufficient numerical robustness, notably with respect to ASTEC applications at full scale conditions (the successful realization by ACAS partners of a large number of plant applications -see WP4.2 below- is a good indicator that this ASTEC V2 reliability objective has been satisfied). The main new capabilities foreseen in the V2.1 version, initially planned in autumn 2010, were mostly covered by the V2.0 successive versions released in 2011 and 2012: full applicability to all situations in Gen.II or III NPPs such as shutdown states or vessel external cooling, except models of air ingress into the vessel after its rupture that is not yet operational. In a similar way, the main new capabilities foreseen in the V2.2 version, initially planned at the end of 2011, were mostly covered by the V2.0 latest revisions: applicability to the major part of sequences in BWR and CANDU NPPs since the V2.0rev3 is already fully applicable except core degradation where some assumptions are necessary (as shown by recent IRSN calculations of the Fukushima accidents).

WP4.2: ASTEC code assessment (ACAS)

The ACAS activity aims at covering a broad matrix of ASTEC reactor applications for the most important accident scenarios in 5 types of reactors (PWR, BWR, VVER, EPR, and CANDU). In complement to these reactor applications, the ASTEC assessment against experimental data is being continued.

ASTEC validation vs. experiments

The main contributors to the ASTEC V2 assessment vs. small-scale experimental data were CIEMAT, ENEA, GRS, IRSN, IVS, JSI, LEI, NUBIKI, RUB-LEE, UJV and USTUTT. Other contributions to this validation came from CEA, EDF, INR, INRNE, KIT, TUS, UNIPI and VTT.

A first synthesis of the ASTEC V2 assessment was released in December 2010 (SARNET2-D4.3 deliverable) and a second one in May 2013 (SARNET2-D4.5 deliverable). The partners have assessed the V2.0 version and subsequent revisions vs. more than 50 experiments in the following facilities:

- On primary circuit thermal-hydraulics: BETHSY 9.1b and 5.2c, PACTEL ISP-33 (the latter for VVER),
- On early phase core degradation: without reflooding, CORA-W2, and with final quenching, CORA-13, QUENCH-03, -12 and -14,
- On late phase core degradation: Phébus FPT4 (debris bed melting),
- On corium behaviour in vessel lower head: LIVE-L1, -L3, -L6,
- On vessel melt jet fragmentation into the lower head : FARO-L-14 and L-18,
- On vessel lower head mechanical behaviour: OLHF-1,
- On release of fission products from fuel: many tests among the whole VERCORS programme,
- On transport of fission products/aerosols in circuits: TUBA TD07, FALCON-18, STORM SR11, POSEIDON-II (aerosol retention in pools),
- On DCH: ANL-IET1RR and U1B,
- On transport of fission products/aerosols in containment: five KAEVER tests, LACE LA4, MARVIKEN Blowdown-16, VANAM-M3,
- On thermal-hydraulics in containment: MISTRA MASP1, PANDA T9, T9bis and T25, PPOOLEX (on condensation in BWR containments), NUPEC M7-1, PACOS Px2.2 (spray effect in German PWR containment),
- On hydrogen combustion in containment: HDR E12.3.2, THAI HD-12 and HD-22,
- On iodine behaviour in containment: SREAS-01, ACE RTF-3B, five Phébus RTF tests, CAIMAN 97/02, and in multi-compartment containment THAI Iod-11 and Iod-12,
- On MCCI: CCI2, CCI-3, CCI-5 (ANL experiments), VULCANO VBU5 and VBES-U2, COMET-L1,
- On vessel external cooling: RESCUE-2,
- On integral tests with prototypical materials: Phébus FPT1 and FPT3, LOFT-LP-FP2,
- On the TMI-2 accident.

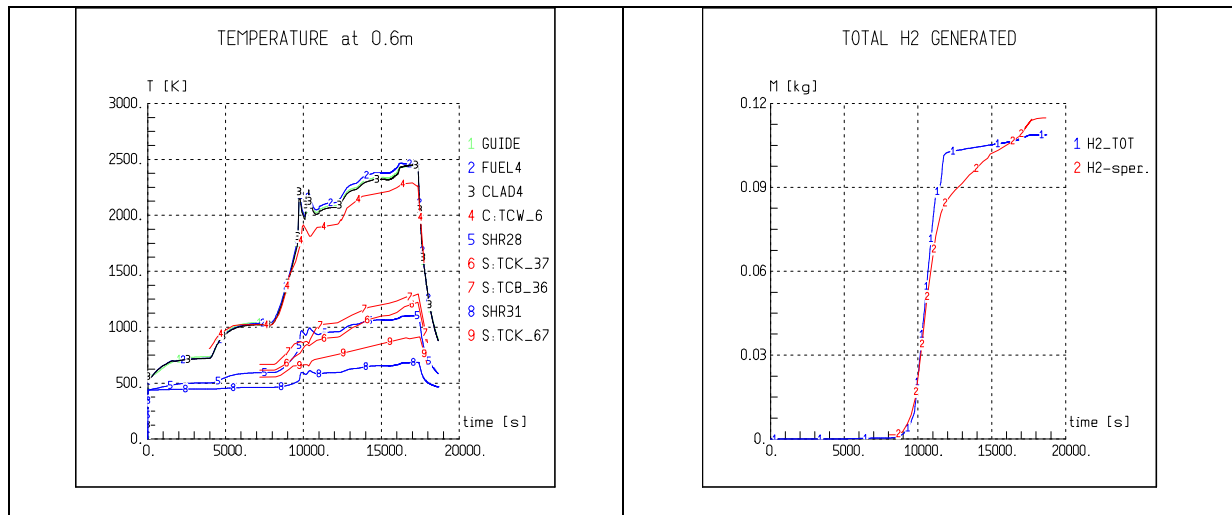


Figure 2: Example of ASTEC V2 validation vs. Phébus FPT3 – Bundle temperature at 0.6 m elevation (left); Cumulated hydrogen production (right)

The Figure 2 illustrates the good agreement of ASTEC V2 results with the trends of Phébus FPT3 experiments for core degradation, in particular with a very good capture of hydrogen production kinetics and slight underestimation of the total hydrogen mass.

In addition, few partners performed also a complete analysis of the Phébus FPT1 and FPT3 integral tests with several ASTEC coupled modules. Besides, one may also underline the successful realization of a probabilistic and sensitivity analysis (using the ASTEC coupling with the IRSN SUNSET tool) in the peculiar frame of the ICARE validation vs. Phébus FPT1. The influence of model parameters to the maximum temperature and the amount of generated hydrogen was evaluated comparing the rank correlation coefficient values.

As general conclusions, good results were obtained on circuit two-phase thermal-hydraulics, core degradation (early phase, debris bed behaviour, corium behaviour in lower head, vessel mechanical failure) except in case of late quenching, release of fission products (except from molten corium pools), thermal-hydraulics of vessel external cooling, containment thermal-hydraulics, hydrogen combustion, aerosol behaviour and iodine behaviour in containment. The agreement was reasonable on MCCI where the models are now at the state-of-the-art and must wait for further improvements of knowledge, and on fission products transport and deposition, where the crucial importance of gas phase chemistry has been underlined.

The main modelling efforts should focus in priority on the reflooding of degraded cores, in particular the corresponding hydrogen production. Using the debris bed and magma recent models, the overall response of ASTEC V2 is rather satisfactory regarding the simulation of thermal-hydraulic behaviour, core melt progression and hydrogen release before reflooding, but it is not the case during the reflooding phase of a degraded core, with a large underestimation of hydrogen production, which is still at the moment a common difficulty for all integral codes because of the general lack of adequate models. Other efforts must notably address MCCI coolability aspects.

ASTEC plant applications and benchmarks

The main contributors to the ASTEC V2 full scale analyses were AREVA NP SAS, EDF, GRS, INR, INRNE, IRSN, IVS, NUBIKI, TUS, UJD SR, UJV, VUJE and BARC. In addition, some other contributions came from EI, TRACTEBEL and VTT for Gen.II PWRs, ENEA and KAERI for Gen.III PWRs and KTH for BWRs. Besides, spent fuel pool (SFP) accidents have been also addressed (preliminary analyses) by LEI and NUBIKI.

Regarding reactor applications, two main categories of ASTEC V2 calculations were carried out: classical ASTEC V2 plant simulations (once-through calculations) and more focussed analyses specifically addressing one key safety issue.

As to the first category, plant applications on diverse NPP types (PWR, VVER...) were performed assuming various kinds of initiating events (cold leg break LOCA, hot leg break LOCA, SBO, TLFW and SGTR). More than 30 different Gen.II plant applications were therefore simulated with ASTEC V2, often focusing on the in-vessel phase but some were extended to MCCI and/or source term evaluation. Number of these ASTEC V2 calculations have been compared with the equivalent ones performed using other reference codes (such as RELAP5, CATHARE and ATHLET for the thermal-hydraulics front end phase, ATHLET-CD for the core degradation phase, COCOSYS for the containment behaviour, CORQUENCH for MCCI processes and of course MAAP and MELCOR integral codes for the whole SA sequence, ...) and reasonable agreement (or even sometimes very good agreement) could be generally obtained. Besides, as to Gen.III PWRs, while EPR analyses have been intensified at IRSN, promising results have been obtained on AP1000. Finally, as to the upgrade of VVER-440 reactors that are in operation in Central Europe, ASTEC V2 could be successfully used to demonstrate the efficiency of proposed plant modifications and adopted Severe Accident Management (SAM) strategies.

The main “classical” ASTEC V2 simulations (once-through calculations) were the following ones:

- TLFW, LOOP, MBLOCA, LBLOCA and SBO sequences in PWR 900 MWe, with benchmarks with MAAP4 code,
- TLFW sequence in French 1300 MWe PWR, along with a detailed sensitivity analysis on final source term release using the ASTEC/SUNSET coupling,
- TLFW and SBLOCA sequences in Konvoi 1300 MWe PWR, with benchmarks with the MELCOR and ATHLET-CD codes,
- LBLOCA, MBLOCA and SBO sequences in VVER-440, with benchmarks with MELCOR,
- SBLOCA and SBO sequences in VVER-1000, benchmarking with the RELAP5 code,
- LOCA sequence (limited to front-end phase thermal-hydraulics) in a 220 MWe PHWR, benchmarking also with RELAP5,
- MBLOCA sequence in AP1000 with benchmark with NOTRUMP code.

As for the core degradation phase, number of partners have adopted during the last SARNET2 period the advanced 2D magma model (instead of the initial 1D candling model), thus following the recommendations from IRSN for full-scale safety analyses. As first conclusion, it has to be emphasized that several partners pointed out that the ASTEC V2 results obtained with the latest V2.0-rev2p2 version are considerably improved compared to earlier calculations. Besides, this conclusion is also confirmed by comparison with other codes; indeed, the benchmark results allow concluding on a globally good agreement between codes for in-vessel and ex-vessel SA progression, despite some differences in some of the studied scenarios on timing of progression and on hydrogen production. The following Figure 3 illustrates a benchmark between ASTEC and MAAP codes, performed by AREVA NP SAS on a French PWR 900. The scenario is a total loss of steam generator feed water (TFLW), with loss of safety injection and opening of the Pressurizer Safety Relief Valves at core outlet temperature of 330°C and with unavailability of the emergency feed water and containment spray. This benchmark allowed also comparison of two corium in-core relocation models: the advanced 2D magma model (recommended by the developers) and the original simple 1D candling model along the rods.

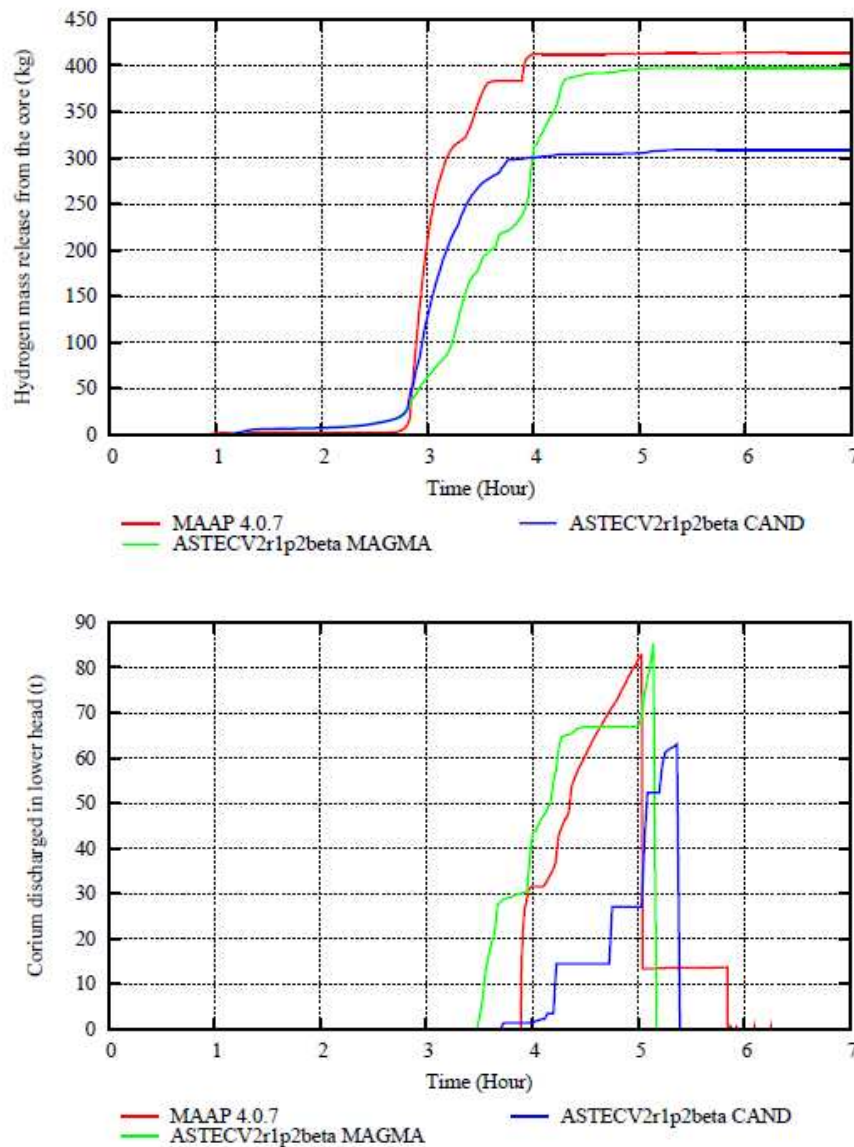


Figure 3: ASTEC-MAAP benchmark on a PWR 900 TFLW scenario - In-vessel hydrogen production (top), corium mass in vessel lower head (bottom)

In addition, about 20 specific safety analyses, focussing onto either a single reactor component or a given physical phenomenon playing an important role in the course of a reactor SA (ASTEC V2 simulations not starting from the initiating event, but covering only a given time window within the whole accident sequence) have been carried out. One must note here that these calculations were activating only a few ASTEC V2 modules together, taking benefit of the high modularity of the code. These topical analyses addressed the following phenomena:

- In-depth analysis of corium behaviour in the vessel lower head,
- Evaluation of IVR concept for VVER-440 and APR-1400,
- Analysis of MCCI processes for PWR, VVER-1000 and CANDU,
- Source term assessment for 1300 MWe PWR and CANDU, and iodine assessment for VVER-1000,
- Hydrogen combustion in CANDU-6 containment,

- Hydrogen risk control (focus on recombination process) in VVER-440.

It is worth to note here that all these studies, addressing very different topics, could be successfully completed thanks notably to the very large flexibility of the ASTEC code.

Finally, the analysis with ASTEC V2 of a SFP dry-out accident applied to the Ignalina RBMK NPP assuming operator actions (such as reflooding by water) could be performed, with a benchmark with ATHLET-CD GRS code in addition. This task has proved the global ASTEC V2 capacity to reasonably simulate SFP accidents.

WP4.3: ASTEC model extension (AMEX)

The AMEX activity aims at extending the current ASTEC applicability to BWR and CANDU reactors. Six organizations were involved: GRS, INR, IRSN, BARC, KTH and USTUTT (the two last ones through a few technical specifications).

A first achievement was reached with a progress report on the adaptation of ASTEC V2 to BWR and CANDU (Deliverable SARNET2-D4.2), mainly focussing on needs of model improvements and corresponding specifications. Several calculations confirmed that most phenomena could already be simulated adequately for such kind of reactors with ASTEC V2.0, except for core degradation models.

Then IRSN has successfully completed the programming of the adaptations of the ICARE module to account for specific BWR and PHWR core components (such as canister walls and cruciform control rods for BWR and sub-channels flows for both BWR and PHWR). This significant evolution, which covers both the thermal-hydraulics aspects for both BWRs and PHWRs (multi-coolant flows) and the degradation aspects for BWRs (multi-paths for vertical downward corium relocation due to the presence of canisters), is now entirely integrated in the future ASTEC V2.1 version under development.

The main other programming evolution was the development by BARC of dedicated ASTEC physical models for the PHWR single channel early degradation phase. In particular, new models for simulating the pressure tube thermal creep deformation have been developed and integrated yet in the ICARE module of the future V2.1 version. Besides, the BARC activity addressed also the thermal-hydraulics in the Primary Heat System, including in particular the elaboration of a typical 220 MWe PHWR system which has been then used to support a benchmarking with RELAP5.

At the end of the SARNET2 project, a very significant progress has been achieved with this ASTEC multi-channel version under development (task carried out through the delegation of one BARC expert at IRSN in the frame of WP2 Mobility programme). With this version, BARC successfully modelled the horizontal core of CANDU/PHWR and coupled it with the primary and secondary heat transport system. The asymmetry in the temperature due to horizontal orientation of channel and under contact between pressure tube and calandria tube was well captured by the new module developed by BARC. This shows that the future V2.1 version will be able to simulate both the Design Basis and Limited Core Damage Accidents scenarios of CANDU/PHWRs.

For BWRs, calculations by GRS confirmed firstly that the CESAR module in ASTEC was applicable to the SA front-end phase, i.e. up to the beginning of the heat up of the rods. The coupling between CESAR and CPA modules has been elaborated with connections between the RPV and the containment and the transient of a total loss of power has been successfully simulated. A comparison was performed between CPA and COCOSYS thermal-hydraulics calculations on a BWR sequence for typical BWR-72 containment. Deviations of results have been identified, which led to determine their origins mainly on differences in the sump heat transfer models. The new multi-channel modelling and the new components representing canisters in the core will of course be very useful

for BWR core degradation calculations to be simulated with the future V2.1 major version, while also allowing making more relevant the future ASTEC simulations of SFP accidents.

1.3.2 WP5: Corium and debris coolability

The overall motivation of the work performed within the WP5 was to reduce or possibly solve the remaining uncertainties on the possibility of cooling structures and materials during SA in the reactor core, in the vessel lower head or in the reactor cavity, so as to limit the progression of the accident. Nineteen partners from European Member States and Associated States (IRSN, AEKI, CEA, RSE, ENEA, KIT, GRS, INRNE, IVS, KTH, LEI, PSI, RUB, THERMODATA, TUS, UJV, USTUTT, NUBIKI, VTT) participated actively in this WP5 and three partners from Third Countries (KAERI, AECL and BARC) contributed too.

Different tasks aimed at improving the understanding of governing phenomena associated with core reflooding and coolability and focus on development and integration of validated models in SA codes like ASTEC in order to reduce the uncertainties in evaluating in- and ex-vessel coolability during different phases of SA progression:

- A first goal was to obtain data on possibility of cooling the degraded reactor core and to turn this data into the knowledge that can be used to support accident management (AM) measures.
- A second goal was to progress towards the closure of SA issues related to in-vessel melt retention. The specific objectives were to create and enhance the database on corium behaviour in the lower head, to develop and validate the models and computer codes for simulation of in-vessel melt pool behaviour, to perform reactor-scale analyses for in-vessel corium coolability and to assess the influence of SAM measures on in-vessel coolability.
- A third goal was to address R&D on coolability of corium in a water-filled cavity.
- A fourth important goal was the transfer of knowledge and modelling tools to existing and future NPPs.

All these tasks included both experimental and modelling activities.

After the Fukushima accident that highlighted the vulnerability of nuclear fuels stored in spent fuel pools (SFP), it was decided to launch an additional sub work-package WP5.5 on this subject.

A joint OECD/NEA/CSNI-SARNET workshop about in-vessel corium and debris coolability was held at the OECD/NEA headquarters in Paris in October 2009. Gathering around 80 participants from diverse organisations, it provided a forum to review the state of knowledge in the area, and to discuss remaining needs and direction of on-going and future R&D work. Four sessions covered general studies, experimental work, phenomenological and modelling work, and specific reactor studies.

Fifty-four papers were presented at international conferences and eighty-five papers were published in scientific journals and/or technical reports. Nine WP5-related PhDs were started in partners' organisations.

WP5.1: Reflooding and coolability of a degraded core (REFCOOL)

Work within WP5.1 concentrated on the ability of water injection to remove heat from a strongly overheated core, typically when degradation is imminent or has already taken place, and ideally to achieve successful quenching. A risk remains that further degradation may occur before quench is achieved, possibly due to thermal or mechanical stresses on the damaged components, to the extent that quench is impeded. There is also a risk of temporarily higher temperatures, accelerated degradation and hydrogen generation due to increased oxidation. The work included experimental and modelling activities, with strong cross-coordination between tasks. Substantial knowledge now

exists concerning cooling of a large intact, rod-like geometry. Therefore, the main thrust of experimental effort is concentrating on debris bed formation and cooling, in order to fill in gaps in the knowledge relating to cooling of a debris bed, to demonstrate effective cooling modes, and to establish cooling rates and limits. Important knowledge has been obtained in that area: it was demonstrated that debris beds may actually be formed under a thermal or mechanical shock to the oxidized claddings, that the debris may relocate between intact rods and are stopped on the spacer grids and that the penetration of water into such debris beds provides an effective cooling, even for small debris, although the quench front progression is slower than in intact rods.

The sources of empirical data on degraded core reflood have been summarized in the “reflood map” (Figure 4) which showed that major gaps in knowledge still remain in the areas of debris coolability and molten pool behaviour with respect to two main consequences (stop of accident progression and hydrogen production). All available experiments on core quenching are here taken into account (besides the TMI-2 accident): LOFT-LP-FP2 at INEL in USA (marked L), PARAMETER at LUCH in Russia (marked P), QUENCH at KIT (marked Q), CORA in KIT (marked C), PBF at INEL in USA, and CODEX at AEKI (marked X).

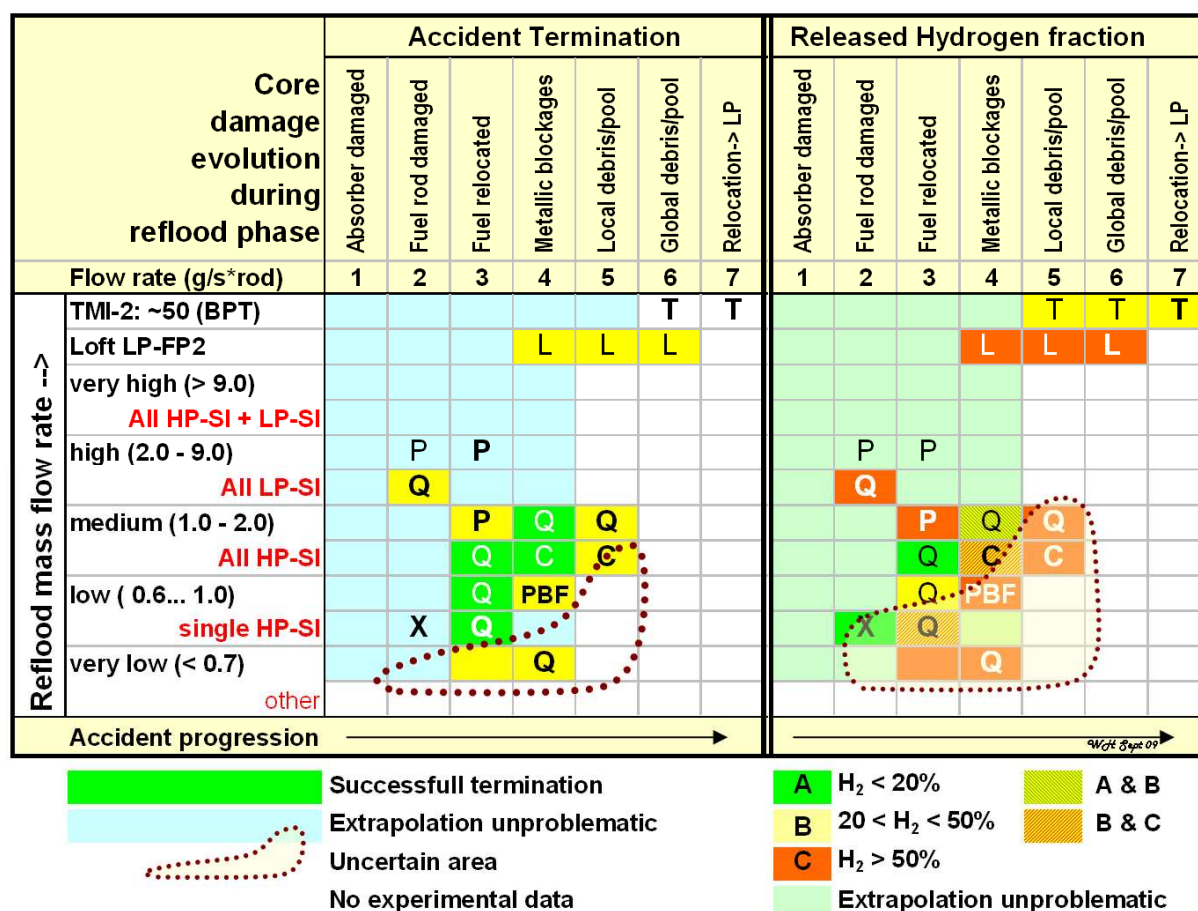


Figure 4: Reflood map based on experiments distributed by available mass flow rate and core damage state for damage progression and additional hydrogen release during core reflooding

The QUENCH-16 experiment performed in the framework of the FP7 LACOMECON project at KIT demonstrated the way in which reflooding can trigger a strong oxidation excursion and degradation following an extended period of oxygen starvation and nitride formation during air ingress. Pre-test planning support for QUENCH-16 was provided by EDF, GRS and PSI. A benchmark on the QUENCH-10 and -16 air ingress experiments was successfully carried out between several simulation codes.

The objective of the very recent debris experiment QUENCH-17 at KIT was to study the in-core debris bed formation and its coolability by water injection from the bottom. Full oxidation of the top part of the claddings was obtained over a length of 500 mm approximately, after a long oxidation phase at a temperature up to 1750 K. During quenching, the claddings were broken, as expected, and a debris bed was formed above the grid spacers located at 350 mm and 1050 mm. The top grid spacer (1350 mm) was not damaged, although it was fully oxidized. The debris bed consisted of pre-fragmented zirconia pellets and large pieces of claddings. This is one of the most unexpected results of the test: cladding fragments have a rather large size (a few cm long) while smaller sizes were expected.

Experiments on the reflooding of fuel rod bundles with partial or full blockage in presence of a bypass line were performed in the CODEX-COOL facility at AEKI. It appears that the bypass line enhances the efficiency of quenching at medium flow rate but seems to reduce it at either low or high flow rate. The experimental results confirmed that a VVER bundle with 86 % blockage rate remains coolable after a LOCA even with the application of bypass line, in the investigated range of parameters.

A large part of the experimental investigation has been performed in the PRELUDE new facility at IRSN, which is providing a large and systematic database on the effect of injection rate, temperature and debris size on the progression of the water front inside a debris bed (Figure 5). Most of the experiments performed in the PRELUDE facility with 60 cm diameter test section delivered the results which were expected at the beginning of SARNET from the planned PEARL facility. One of the findings is the occurrence of 2D and 3D effects during bottom flooding at higher injection rates. The PEARL facility of larger size (110 cm diameter) is near the end of its construction at IRSN and the first experiments are planned in the beginning of 2014, which will extend the database obtained in PRELUDE.

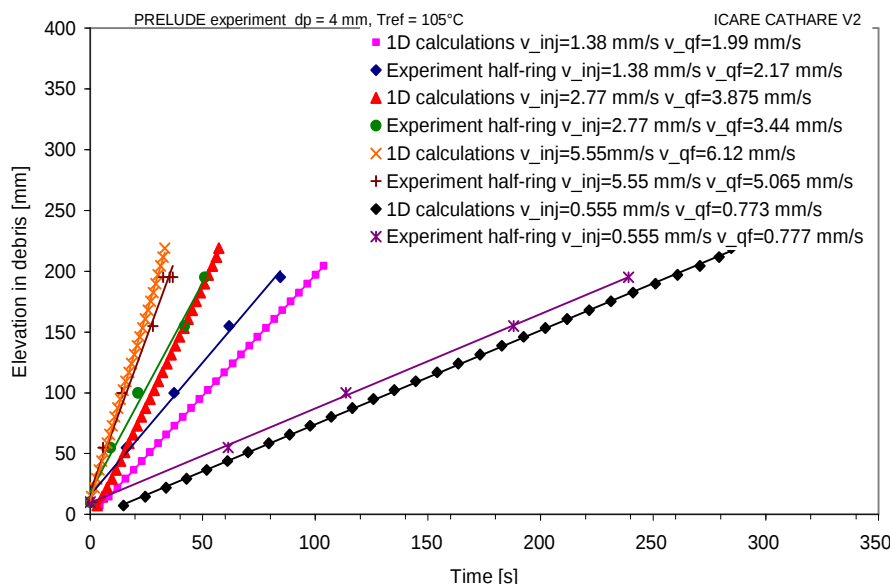


Figure 5: Quench front velocity for 4 mm particles in PRELUDE experiments

DEBRIS analytical tests with debris beds at USTUTT in Stuttgart complement the PRELUDE experiments and include top and side injection tests with irregular particles, higher pressures and temperatures. For poly-dispersed bed configuration the quenching behaviour showed pronounced multi-dimensional features. Due to non-homogeneities of temperature and porosity, penetration of water is facilitated at the periphery of the debris bed. In the tests, typically two phases were observed. In the first phase, water penetrated the bed locally in the form of streaks, which is a relatively slow process. After the water has reached the bottom, establishing a local “water supply

channel”, the remaining larger part of the bed is flooded in a relatively fast manner. Thus both DEBRIS and PRELUDE programs supported the quantification of basic laws to predict coolability behaviour under a wide range of conditions.

Modelling efforts have aimed at assessing and validating the models in system-level and detailed codes for core degradation, oxidation and debris behaviour. The analysed data comprise PRELUDE, DEBRIS, QUENCH experiments and the TMI-2 accident. Extensive analyses have been performed by IRSN using ICARE/CATHARE code to support and interpret the PRELUDE experiments and support the preparation of PEARL experiments, including the effect of the test variables on cooling. Comparison of quench front elevation as a function of time shows that the ICARE/CATHARE model is able to predict quenching velocity for different inlet flow rates and different particle diameters. Additional analyses of PRELUDE data using the USTUTT MEWA code delivered results consistent with IRSN. MEWA analyses of the DEBRIS data were also performed by USTUTT.

A major effort was carried out by ENEA to investigate the effect of many parameters describing the TMI-2 core conditions on the extent of degradation and hydrogen generation in TMI-2, using the ICARE/CATHARE code. Some cases lead to complete cool-down of the core whereas other cases lead to molten pool formation. Criteria for the success of reflooding have been derived. This effort is in conjunction with the OECD/CSNI benchmark exercise BETMI2 being supported in WP5.4.

WP5.2: Remelting of debris, melt pool formation and coolability (MPF)

Work within WP5.2 aimed at investigating corium behaviour and coolability in the LWR lower head during a SA.

A very comprehensive experimental test matrix has been realised in the LIVE-2D and LIVE-3D facilities at KIT. The transient and local thermal loads on the RPV wall under different melt relocation scenarios were examined. The steady-state pool behaviour of both homogenous pool and two-layer pool configurations were investigated. In all LIVE-3D experiments the melt pool temperature profiles, axial and radial heat flux distribution through vessel wall, crust thickness profiles, transient behaviour of melt temperature and heat flux were obtained. Valuable experimental results such as the temperature of crust and boundary layers were obtained for the modelling and analytic of the characteristics of corium with crust formation.

RESCUE-2 tests were performed at CEA to study external cooling of a lower head of VVER-440/V213 type. Three tests relevant to the IVR of the VVER-440/V213 were performed, adapting to the RESCUE-2 geometry the VVER-440 thermal loads for three possible heat flux profiles, as calculated by the ASTEC code. The thermal loads correspond to three tests: SARNET-A with a typical stratified corium pool; SARNET-B with transition from a homogenous pool to a stratified MASCA-type pool; SARNET-C similar to SARNET-A but having a restriction at the outlet. For all three tests the outlet flow is a two-phase flow at the saturation temperature. The highest wall temperature is located at the position where the heat flux is the highest. Concerning the SARNET-B test, during the change of thermal load the vessel temperature increases in the zones where the thermal flux increases and decreases where the thermal load decreases. In the SARNET-C test an oscillating behaviour of the flow was observed when the valve is at an opening position of significant pressure loss. Moreover, the higher is the pressure loss at the exit, the smoother is the flowrate.

At KTH a coupled thermo-mechanical creep analysis was applied to a BWR lower head (Figure 6). The study reveals that if only Control Rod Guide Tube (CRGT) cooling is activated, then two RPV failure modes are possible. These two failure modes may lead to different melt releases in terms of breach size, melt mass and melt compositions and superheat. If the external vessel cooling is implemented right before the creep accelerates, then the analysis confirms the possibility of melt retention by CRGT cooling plus vessel external cooling for all melt pool configurations considered.

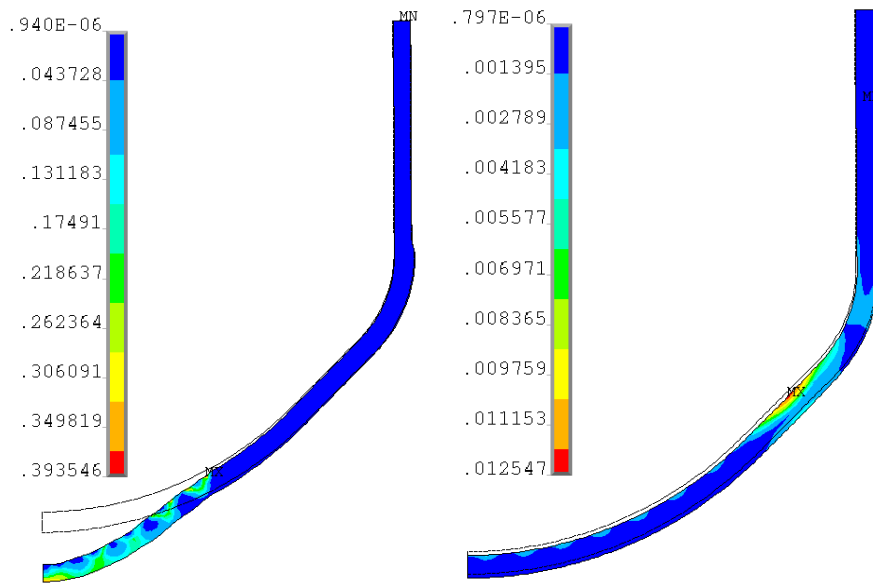


Figure 6: Von Mises creep strains and displacements of the BWR vessel for melt pool depths 0.7 m (left) and 1.9 m (right)

WP5.3: Ex-vessel debris formation and coolability (EXCOOL)

The focus of R&D within the WP5.3 is to get an overall perspective on coolability of melts released from a damaged RPV and relocating into a water-filled cavity. In particular accident management concepts for BWRs with deep water pools below the vessel are addressed, but also shallow pools in existing PWRs were considered, addressing the questions of partial cooling and time delay for MCCI. Considerations on AM measures during Fukushima accident like flooding of the dry well demonstrate the importance of this field.

DEFOR experiments at KTH were performed to study the jet breakup phenomena and debris bed formation processes. Realistic debris are to be considered which concern the breakup process (higher density material, thickness of jet of several centimetres, larger mass involved resulting in a sufficiently long jet, variation of the water depth), yielding a particle size distribution or effective diameters, and the bed build-up during mixing with water and settling of drops under freezing and crust formation, if freezing is not sufficient with formation of liquid and cake parts in the bed. Generally, the experiments aimed at obtaining the shape of debris beds and the internal structure.

POMECO at KTH and DEBRIS at USTUTT analytical tests with debris beds were focused at getting constitutive laws for realistic debris, i.e. local mixtures with irregularly shaped particles of different sizes. Both top and bottom flooding of debris bed were investigated applying volumetric induction heating. Both experimental set-ups aimed at investigating specific multi-dimensional effects e.g. using downcomers with bottom openings and variants of lateral openings. The scale of the experiments differs. While the DEBRIS test section has an inner diameter of 12.5 cm, the POMECO-HT facility uses a rectangular cross-section of 20 cm x 20 cm (height of debris bed in both facilities about 60 cm). The dryout heat fluxes were determined for particulate beds packed with various particles and the effect of coolant injection on dryout heat flux was analysed as well as the effect of downcomers and different bed configurations.

Experiments at VTT in the COOLOCE facility were more directly oriented to reactor situations using heap-like beds of conical shape with a base diameter of 1 m and a height of 0.7 m. The beds are flooded by the top or flooded by the top and laterally, using spherical beads or irregular gravel as debris simulant materials (see Figure 7 for simulation of one experiment with the MEWA code by USTUTT). A geometry comparison concerning the coolability for equal volume and bottom surface

beds suggests a poorer coolability of the conical bed than that of the cylindrical one due to greater height of the conical configuration. The best coolability was found in top and lateral flooded cylindrical beds.

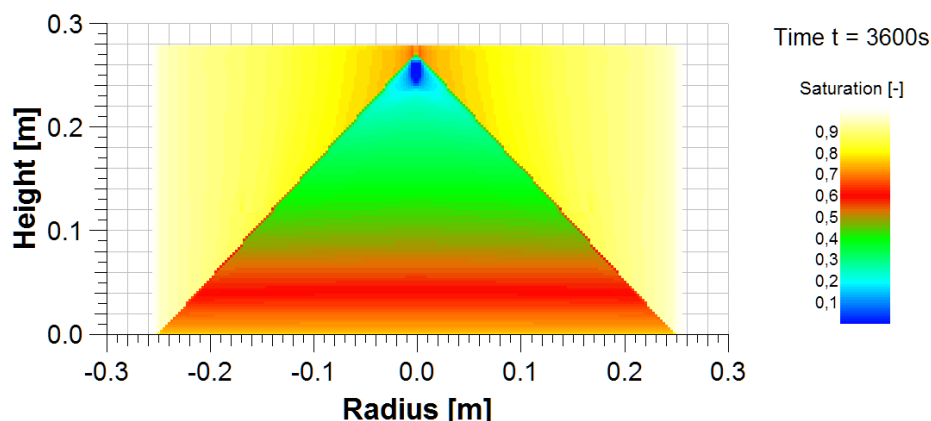


Figure 7: MEWA simulation of the conical debris bed of the COOLOCE experiments, saturation (liquid volume fraction in the pores) in post-dryout conditions

WP5.4: Bringing research results into reactor application (COOL-RA)

Work within WP5.4 comprises applications of different mechanistic and integral codes to different reactor designs. Activities within WP5.4 include modelling work with important inputs from WP5.1, 2 and 3 and with strong links to SAM strategies that are adopted in different NPPs. The work was mainly focused on core degradation, melt relocation to the lower plenum, quenching of corium by residual water, re-melting of debris beds and molten pool formation in the lower head during SA sequences for different LWR designs. Significant attention was paid to in-vessel coolability issue during different accident stages and specifically to stabilization and localization of a volumetrically heated molten pool in the RPV lower head, with application to external reactor vessel cooling. The last point is in general considered as an ultimate goal of SAM. Achieving the in-vessel retention (i.e. stabilized state, when the decay heat from the molten pool is removed through the RPV wall) is a promising SA strategy especially for reactors of smaller size where relatively low thermal loads on reactor wall are expected.

The most important activity within the WP5.4 was the OECD benchmark exercise on an alternative TMI-2 accident scenario. This is a follow-up of the previous OECD benchmark exercise on an alternative TMI-2 accident scenario. This project was being carried out by a group of participants including members from both WGAMA and SARNET. Eleven organizations from 8 countries participate in this exercise with 5 different codes. The objective is to examine three different SA sequences in the frame of a code-to-code benchmark. The impact on hydrogen production, core coolability, corium relocation into the lower plenum and vessel failure was addressed. The codes were able to calculate the accident sequence up to the more severe degradation conditions, which showed a significant increase of numerical robustness with respect to previous similar exercises. The first deviations in the results are registered after the initiation of in-core melt progression and material relocation phenomena, resulting in core geometry change. More important deviations can be obtained in the late phase core degradation, e.g. on the total corium mass in vessel lower plenum. The main origins of these discrepancies are first the modelling differences and, only in a lower extent, the user effect. The highest uncertainties remain for scenarios of late core reflooding, i.e. on a severely damaged core.

WP5.5: Spent fuel pool analysis (SFP)

The Fukushima accident highlighted the vulnerability of nuclear fuels that are stored in SFP before their evacuation and final disposal or possible reprocessing. Several SARNET participants have

already done SFP analyses for various types of reactors. In early 2012 it was decided to start sharing data and experience in order to optimize and reinforce that activity and possibly reach common conclusions and define future R&D projects. The studies have focused on the evaluation of the SA codes capabilities, limitations and needs for improvement. This assessment was based on:

- The state of knowledge, especially with regard to phenomena related to oxidation in air of the fuel rod claddings,
- The code assessment on integral tests like QUENCH or PARAMETER tests (the latter in ISTC projects in Russia) allowing to study accidental transients of oxidation in air of fuel rod cladding, ending by reflooding and SFP tests allowing to study the behaviour of one of several fuel assemblies (FA) for representative transients of loss of coolant SFP accident, inducing fuel claddings oxidation in air and burn propagation,
- The assessment of SFP accident scenarios with different SA codes for a range of SFP geometries, residual power levels and distributions of the FAs.

SA codes have been developed for reactor applications, their physical models and the validation of these models are based on experimental data representative mainly of reactor configurations. Of course, some data and models seem directly reliable for SFP applications, like some analytical tests. But some transposition to SFP accident assessment seems more difficult and uncertain, due to the differences of geometries and of transient conditions. The work carried out in the framework of SARNET2 aimed at an attempt to share and harmonize the SFP transient analysis, using these SA codes as no specific (or adapted) simulation tools exist currently.

The studies have shown that the modelling assumptions in SA codes to represent the SFP geometry (upper part of building, hall, pool, FA pool...) can have a strong impact on the gas flow between the different parts of the building (hall, upper plenum, pool, FAs) and consequently on the evaluation of the risks associated to the transient. They also raise questions about the reliability of some results obtained with these codes, in particular:

- The phenomena related to the cladding behaviour in the presence of air or a steam / air mixture, such as oxidation, nitriding and embrittlement,
- The phenomena of natural convection and boiling in the fuel building. In fact, the conclusions on the coolability of FAs can be very different as a function of the calculations. Some studies suggest, for a complete loss of water transient, that the induced air flow may be sufficient to remove the power, while for other studies this conclusion depends on the air flow that could actually flow in the FAs. But most of these calculations seem to use thermal-hydraulic parameters/models which that are designed and benchmarked for reactor core geometries, and are not necessarily appropriate for SFP geometries. A consequence is that the gas flow might be strongly overestimated and hence non-conservative. The OECD SFP experiments showed ignition in a simulated 3-year old spent fuel element in air.
- The conditions of air ingress in the assembly; according to the water depth, the assembly power, and the intensity of boiling. A few studies show that for some conditions, during the phase of FA dewatering, the air ingress flow through the top of the assembly (counter-current of steam flow) can cool down the FA upper part.
- The coolability of dry fuel assemblies after water injection.

It appears that the SA codes usually used for reactor applications cannot reliably assess the phenomena of natural convection in the different parts of the building and the conditions of air ingress in the assemblies. It has to be mentioned that the transient phase before fuel assemblies dewatering is usually assessed using CFD codes which cannot simulate the degradation and oxidation phases. The recommendations are:

1. To use carefully the current results of SFP transients issued from SA codes.
2. In the continuity of past activities (performed in the frame of SARNET FP6 and FP7 projects on air ingress analyses after vessel rupture for reactor SA), to carry out different types of new experiments in order:
 - o to better estimate the margin before temperature runaway of the oxidation reaction,
 - o to improve knowledge on the role of nitrogen in the acceleration mechanisms of cladding degradation and on the mechanical behaviour of oxidized/nitrided claddings,
 - o to estimate the ability of fuel rods weakened by oxidation/nitriding to maintain their integrity during a spray of water or during handling in the post-accident phase.
3. To continue the analysis of SFP transients with SA codes in the frame of a benchmark on SFP geometry like Fukushima one, in order to define more precisely needs of R&D on large-scale flow convection, impact of partial dewatering or air flow on thermal runaway and fuel degradation.

The results of this benchmark should complete the OECD state of the art report on SFP under way (planned to be released at the end of 2014) and identify research activities required to reduce the uncertainties on SFP accident assessments.

1.3.3 WP6: Molten Corium Concrete Interaction

In the case of a SA with vessel melt-through, the containment concrete is the ultimate barrier between the corium and the environment. Scenarios are considered in this WP where the reactor pit is initially dry but water injection may occur later during Molten Core Concrete Interaction (MCCI). This issue has been ranked by the SARNET SARP (Severe Accident Research Priorities) group with high priority. The case of a pit flooded prior to corium ejection is out of the scope of the WP6 since it is addressed in WP5 on the generic topic of corium coolability.

Twenty-one partners were participating to this WP6: AREVA GmbH, CEA, EDF, EI, GRS, INRNE, IRSN, JRC (ITU), KTH, KIT, NUBIKI, RSE, TRACTEBEL, THERMODATA, TUS, UJV, USTUTT, VTT, PSI, BARC and KAERI (for the last two, participation limited to WP6.4). JAEA expressed interest to join WP6 but was not able to attend the meetings.

These activities have been organized into 4 issues, each with a co-ordinator:

- The effect of the concrete nature on ablation profiles (Michel Cranga, IRSN);
- The role of metallic layer on MCCI (Jerzy Foit, KIT);
- The efficiency of late-water cooling of a MCCI pool (Weimin Ma, KTH);
- Bringing research results to reactor applications (Pavlin Grudev, INRNE).

One of the major outcomes of the joint activities lies with the regular organization of technical meetings (7 throughout the 4 years) where the research teams exchanged in depth. Their interest is such that a large majority of partners decided to pursue these exchanges meeting, on a voluntary basis within NUGENIA/SARNET.

A transverse important activity was the writing of a state of the art report (SOAR) on dry MCCI, including the main findings of WP6.1, 6.2 and 6.4 and released in summer 2013. Fruitful links have been established with the MCCI SOAR OECD/NEA project which has been decided later. Exchanges of draft sections have enabled a better homogeneity between the two documents which have different scopes since there is an important focus on coolability in the OECD SOAR.

Some main lessons on MCCI in dry conditions were drawn:

- first on the origin of discrepancy between the isotropic ablation of limestone-rich concretes and the larger lateral ablation than vertical ablation of silica-rich concretes, mainly due to the structure of pool/concrete interfaces;
- secondly for MCCI in oxide/metal pools, phase repartitions which are different from simple-layer assumptions considered in MCCI codes on the basis of VULCANO experiments and, in MOCKA experiments, a stable stratification but with a density contrast between phases typical only of the long term MCCI phase, as well a strong effect of iron bars in concrete leading to a more isotropic ablation;
- thirdly reactor applications showing no pool stratification with metal below in case of Limestone Common Sand (LCS) concrete and a late basemat melt-through exceeds 8 days but, in case of siliceous concrete, possible stratification and basemat melt-through only after a few days (but many uncertainties remain in the latter case such as assumption of prevailing lateral heat transfer as observed in experiments or increase of water content in concrete that might suppress pool stratification and delay the basemat melt-through very significantly).

The SOAR allowed also identifying the remaining uncertainties on MCCI in dry conditions. They concern the multidimensional convection within the corium pool and the prediction of the thermal resistance of the pool/concrete interfaces in the long term phase in particular in case of siliceous concrete, the existence and stability of stratification of an oxide/metal pool and, if any stratification, the induced multidimensional ablation in a situation involving a concrete with iron rebars.

WP6.1 Effect of the concrete nature on ablation profiles (ABPROF)

One of the unexpected results of the 2D MCCI experiments (e.g. at ANL and CEA) is the fact that while limestone-rich concretes exhibit an isotropic ablation, silica-rich concretes are more ablated laterally than axially (so-called anisotropic ablation). The causes for this anisotropy are not yet fully understood. The major objective of this work-package is to understand this phenomenon.

Most of the previous prototypic corium-concrete interaction tests were in a geometry (called 1D) in which there was only one concrete wall (usually the bottom wall), made of ablatable concrete, the others being refractory. Recent MCCI experiments performed within SARNET2 (VULCANO at CEA) and OECD/MCCI (CCI at ANL) projects studied geometry (called 2D) with a horizontal concrete wall and at least a vertical concrete wall. As an example, Figure 8 presents the ablation profiles from two “anisotropic” VULCANO tests.

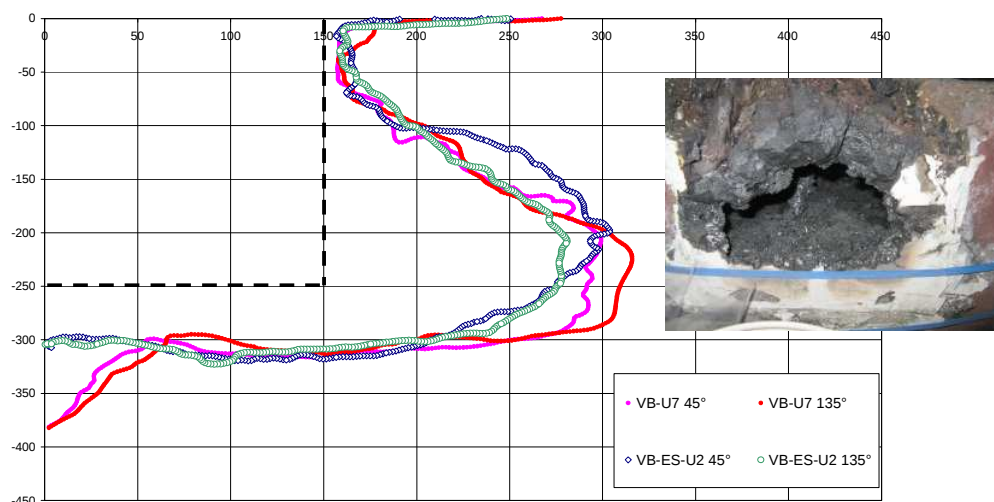


Figure 8: Example of anisotropic ablation profiles during VB-U7 and VB-ES-U2 experiments (Insert: post-test view of the VB-U7 cavity)

An analysis of previous experiments showed that during the 1D experiments with both types of concrete and the 2D experiments with limestone concrete, the pool temperature followed more or less the evolving pool liquidus temperature, whereas, for 2D tests with siliceous concretes, the pool was largely below liquidus temperature and the ablation was anisotropic.

In the VULCANO facility at CEA Cadarache, it has been decided to launch a series of separate-effect tests with prototypic corium and “artificial” concretes in order to determine which of the differences between siliceous and limestone concretes controls the ablation shape. The first test was performed on June 18, 2009 with a concrete made with cement and clinker aggregates, i.e. limestone without carbonates. Then the VBES-U3 VULCANO test was performed in 2010. The test section was made of siliceous mortar (i.e. concrete without coarser gravels) instead of a complete concrete, in order to assess the effect of large aggregates on the anisotropy. Even though the same power per volume as in previous test has been injected, very limited ablation was observed. This may be due to the high viscosity of molten mortar: although it has the same composition as the concrete used in a previous test, it appears that quartz (silica) gravel does not melt easily and thus there was a less siliceous melt in previous tests. The separate-effect test VB-ES U4 was performed by CEA in July 2012 with oxidic concrete: it was devoted to the study of the role of the initial crust at the bottom interface. The test used metallic zirconium at the bottom of the concrete cavity as a heat source in order to limit the growth of a transient thick crust at the bottom corium-concrete interface. About 24 kg of corium (with a similar load as for previous tests VBES-U2 and VBES-U3) have been poured in the cavity at 2180°C. The concrete cavity was only ablated in the axial direction (downwards). After a first ablation, induction coupling was apparently poor and ablation stopped. No metallic zirconium has been found during the post-test examinations, confirming that the zirconium was indeed oxidized and mixed with the oxidic pool. The pool sample was quite homogeneous. Its composition was slightly enriched in concrete decomposition products and the Zr/U ratio was increased compared to the initial corium.

It has been decided to use the results of two oxidic tests VB-U5 (silica-rich concrete) and VB-U6 (limestone-rich concrete) for a benchmark on MCCI codes. Ten participants took part in this work. Six different codes have been used to compute the main phenomena during the interaction between real concrete material (siliceous-rich and limestone-rich concretes) and molten prototypical corium. Clearly codes imposing the anisotropy (by fitting a multiplicative factor applied to the convective heat transfer coefficients or imposing values of thermal resistances at pool/concrete interfaces) or using heat transfer models leading intrinsically to anisotropy permit to get a better agreement for the final cavity shape (Figure 9). It must nevertheless be noted that VTT calculations with CORQUENCH provided a good shape without explicitly choosing an anisotropic ablation.

The main conclusions of the benchmark are the following ones:

- There are many similarities in the predicted trends, in the ranges accounted. Nevertheless some major differences between modelling approaches can be observed.
- The ablated volume is controlled by the ablative energy, thus by the amount of energy radiated through the upper surface which depends on the code heat transfer models (heat convection distribution and interface structure) and also on the upper crust interface temperature; moreover most codes overestimate the ablated concrete volume if significant conduction heat losses through the concrete are not taken into account especially in case of VB-U6.
- Assuming an interface temperature around or slightly below liquidus provides good estimates of the pool temperature whereas the other models give large discrepancies of several hundreds of Kelvin at least in the initial MCCI phase of the VBU5 experiment; however later the calculated temperatures cannot be compared with the experimental data since the overall pool temperature evolution is not measured.

- Cavity shape are rather well predicted: the VB-U5 with siliceous concrete required taking into account anisotropy, either explicitly (with MEDICIS, TOLBIAC) or implicitly (with CORQUENCH and WECHSL); axial ablation is generally overestimated.
- All codes underpredict the void fraction (even with limestone test U6); the reasons are probably that the drift flux model is not valid in a 2D gas injection situation, even if different approaches are used in codes to adapt 1D experimental data to a 2D configuration with gas sparging from two perpendicular interfaces.
- Crusts, if they exist, will have a composition close to the current pool composition.

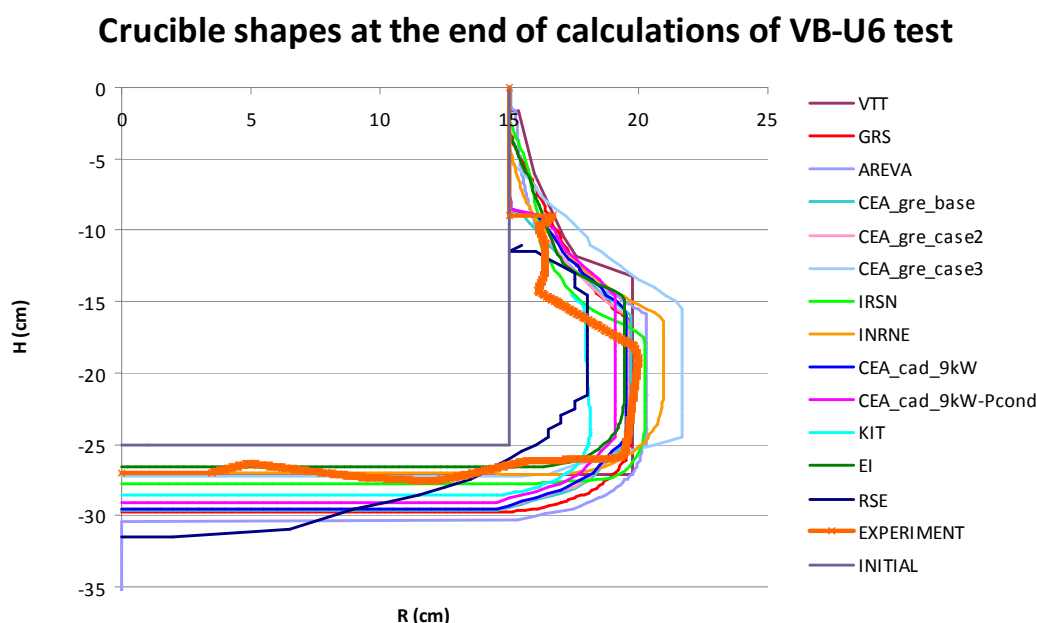


Figure 9: Comparison of experiment and benchmark calculations for test VB-U6

Data from previous VULCANO experiments and from the EC-funded VULCANO VB-U7 experiment with EPR sacrificial concrete have been provided to partners for interpretation.

Up to now, it is still not possible to propose a comprehensive modelling of MCCI that could predict the observed anisotropy and all the parameters of the experiment. But it must be reminded that we are using multi 0D quasi-steady state modelling to model an intermittent ablation process with a complex geometry both at the interface and a complex convection pattern in the pool because of combined effects of gas bubbling and solutal convection. However, the interpretation of 2D MCCI experiments permits to propose some improved models better validated against the experimental database compared to those of a previous work, although the models are still of parametric type. For instance, IRSN applied the latest versions of ASTEC/MEDICIS on VULCANO VB-U5 and VBES-U2 tests, as well as to the CCI and VB-U6 tests, which led to propose a new set of assumptions and models for homogeneous pools. GRS has continued the work on proposal of effective heat transfer coefficients for MCCI codes, based on CFD calculations and taking into account the results of recent MCCI experiments.

To provide insights to the thermal-hydraulics of a MCCI pool, the CLARA experimental programme with low temperature simulant fluids has been launched at CEA Grenoble. The objective was to measure convective heat transfer coefficient on the lateral and bottom isothermal walls of a pool percolated by air simulating the release of gas generated during MCCI. The aim is to improve the database for validating 2D convective heat transfer correlations which are used by MCCI codes. The pool dimensions are 0.5 m long by 0.25 m wide by 0.25 m high. Vertical front and back walls are

made of transparent Plexiglas and allow flow visualisation. They are equipped on their inner face with copper electrodes which allow direct heating of the pool by dissipation of electrical power. For tests without gas injection from the side, the lateral to bottom heat transfer ratio is close to 1 for low fluid viscosity. This ratio decreases for fluid with high viscosity. As for tests with gas injection both from bottom and lateral walls, the temperature in the pool is homogeneous; in case of a low fluid viscosity (below 0.025 Pa.s) the ratio of lateral convective heat coefficient to axial convective heat coefficient is higher than 1, whereas in case of a higher fluid viscosity above 0.025 Pa.s, the ratio of lateral convective heat coefficient to axial convective heat coefficient is smaller than 1. For tests without gas injection from the bottom, the lateral to bottom heat transfer coefficient ratio is much larger than 1 even for viscous fluid and a significant temperature gradient appears in the pool. However, the vertical temperature profile remains less inhomogeneous and the lateral to bottom heat transfer coefficient ratio is smaller than for natural convection. During the year 2012, the second experimental campaign with simulant materials in the CLARA facility was launched with a larger pool dimension: 2 m long by 0.25 m wide by 0.25 m high. This configuration permitted to investigate a more realistic aspect ratio h/L for MCCI. CLARA experiments provide a comprehensive void fraction data base, which covers all bubbling regimes possibly expected in the reactor case with lateral and horizontal gas injection. The proposed model is mechanistic and is a generalisation of the Zuber and Findlay's model in other bubbling regime than churn-turbulent regime without assuming the bubble diameter is obtained from Laplace's equation and taking into account the impact of lateral gas injection. This model will be simplified to be used in ASTEC (in the lumped parameter MCCI MEDICIS module) and also applied at the experimental results of the second campaign when available.

Smaller scale real material experiments have been conducted:

- SICOPS by AREVA NP GmbH on interaction of mixed oxide-metal concrete with silicate concretes as well as of oxidic corium with concretes. They showed that ablation/heat flux correlations were identical in 1D for classical siliceous concrete and for the EPR ferro-siliceous sacrificial concrete,
- COMETA by UJV on thermochemistry. These tests were dedicated at studying corium-concrete melts with the view of improving the NUCLEA database,
- Experimental work at JRC/ITU with laser heating was performed to provide new data on phase diagrams that will be introduced in the NUCLEA database.

In support of the models coupling thermal-hydraulics and thermochemistry, the NUCLEA thermodynamic database of the THERMODYNAMIC company has been improved on the following issues:

- Addition of new experimental results to the database. The Ba-O-U, Mo-U, Mo-O-U, B-Fe-U and B-Ni systems have been reviewed ;
- Analysis of the VULCANO VB-U7 (HECLA EPR concrete) test: it seems that the calculated (Gemini + NUCLEA08-1) temperature for the liquidus of the concrete is 270°C higher than the one experimentally measured. This point has been analysed and it appears that there is a very stiff variation of liquidus temperature around the composition of this concrete, explaining the observed discrepancy which remains within the uncertainties (in terms of composition);
- Continuous survey of the scientific literature about the critical assessment of the chemical systems involved in NUCLEA has been made to ensure a good validation and qualification of the database.

WP6.2 Role of metallic layer on MCCI (METLAY)

The first task was limited to single phase oxidic pools. Actually corium is made of two phases with a miscibility gap. Two configurations are considered in the models: either the two phases form an emulsion, which is assumed to behave as a homogenous equivalent fluid, or there is a gravitational stratification (the oxide becomes lighter than the metal due to the introduction of light concrete oxides). In this last case, the heat transfer between the oxidic layer (where more than 90% of the decay heat is produced) and the lower metallic layer is larger than the heat transfer at the oxidic layer sides. Reactor scale calculations indicate that the major uncertainty lies on the stratification thresholds: ASTEC calculations of a typical reactor sequence give a basemat melt through between 2 and 6 days, depending only on the choice of threshold correlation.

Nowadays, two correlations exist for the heat transfer coefficient at a liquid/liquid horizontal interface. A more accurate correlation has to be established and this work is a first attempt to do so using Greene's experimental results and the results of a recent program with simulants, called ABI, launched at CEA Grenoble. Two test series were performed. The first one was carried out with water or different oils over Wood's metal, and the second one with gallium instead of Wood's metal. It indicated that heat transfer depends on properties of both liquids and of bubble sizes. A new correlation depending on bubble sizes has been fitted on the latest results as on earlier Werle and Greene data.

A series of experiments have been performed in the large-scale MOCKA (KIT) facility to study the interaction of a simulant oxide and metal melt in a stratified configuration. To allow for a long-term MCCI, additional enthalpy was supplied by alternating additions of thermite and Zr. Heat generated by the thermite reaction and exothermal Zr oxidation reactions was mainly deposited in the oxide phase. Siliceous concrete crucibles were used with an inner diameter of 25 cm using initially 39 kg of Fe and 3 or 4 kg Zr, overlaid by 70 kg oxide melt (Al_2O_3 , CaO). The melt temperature at start of interaction was approximately 2173 K. The long-term axial erosion by the metallic phase became more pronounced and was a factor of 2-3 higher than the lateral ablation. This is in agreement with results obtained in the former BETA and COMET-L experiments. In contrast to the findings in the latter experiments, significant lateral concrete erosion by the oxide melt was observed. The more pronounced downward erosion seems to be inherent to the erosion by metal melts. One of the still unresolved issues is the long-term interaction of a melt with a reinforced concrete. New experiments were performed: two with iron reinforcement bars (called "rebars") have shown an almost isotropic concrete ablation (Figure 10) while tests without rebars showed a preferential axial ablation.



Figure 10: MOCKA 3.1 - View of the rebars before concrete pouring (upside-down) and post-test cut of the test section

Post-test analyses of past VULCANO Oxide-Metal experiments indicated that it is quite difficult to achieve a stable stratified configuration. A new test was performed by CEA with conditions favouring stratification. The VULCANO VBS-U4 test has been performed to verify whether the non-horizontal phase segregation between metal and oxide observed in the previous VBS-U2 and VBS-U3 experiment was due to the density ratio between phases in this test. Therefore an initial oxidic load composition with significantly more light oxides has been chosen. The metallic phase has been found not only as a bottom layer but also as vertical “tongues” on the vertical concrete walls (fortunately, it did not extend on the refractory inert wall, which would have prevented induction coupling). CEA performed the post-test examination of Oxide-Metal tests VBS-U3 and U4 made with corium + steel + siliceous concrete and two different initial oxide compositions (10-35 wt% concrete). This phenomenon seems to be generic but is not understood yet.

WP6.3 Efficiency of Late Water Cooling (LWC)

Water injection on top and/or bottom of a corium pool in the cavity is the main available mean to terminate the concrete ablation in Gen.II reactors. Recently, interest has been to pursue R&D on concepts that could be used to provide bottom-cooling in the pit of current reactors. For new reactors (e.g. EPR), this has been realized by specific designs of the reactor cavity. Experiments have been performed to investigate the efficiency of water cooling of corium in the reactor cavity through top flooding, e.g. in the OECD MCCI project, and/or bottom injection. This sub-work-package aims at research of potential back-fitting options for MCCI mitigation, with the ultimate goal to predict the efficiency of water cooling, and to realize ex-vessel corium coolability.

The joint interpretation of the water cooling tests performed in OECD-MCCI program (e.g. SSWICS1-13, CCI-6) and VULCANO-COMET test was performed.

USTUTT performed development, validation (on SSWICS tests) and application of model and code (MEWA-COMET) for the coolability analysis of a melt layer under bottom and/or top flooding, including the effects of MCCI gas production.

For top flooding, the following factors are considered:

- Freezing at top with subsequent crack formation in crust, ingress of water through cracks versus counter-current flow of steam (and MCCI gases), dependence of freezing combined with crack formation on water inflow through already formed crust, attenuation of effect with increasing depth of crust.
- Entrainment of melt into flow of MCCI gas out of holes in the crust, leading to formation of porous debris on top of the crust.
- Porosity formation in melt layer due to sparging gases, perhaps in combination with water ingress from top.

The results show that sustained crack formation and practically constant crack formation and cooling by water ingress would lead to plateau and rapid reduction in heat flux.

IRSN performed a review of available top quenching models both for melt ejection hydrodynamics and for determination of the geometry of ejection holes build-up through the upper crust, as well as proposals for modelling improvements. GRS provided a simplified top flooding model for use in ASTEC by simulation of heat transfer from the molten MCCI pool to an upper water layer in conjunction with the use of effective heat transfer coefficients at the pool boundaries. The results are consistent to the SSWICS experimental data.

KTH calculated the CCI tests performed in the OECD/MCCI-1 and MCCI-2 programs with the CORQUENCH code, with good agreement on CCI-2, CCI-4 and CCI-5 but not on CCI-3 and CCI-6 tests. This was also employed to investigate the coolability of corium in a PWR cavity by late water

injection, but results showing a full corium quenching were contradicting MELCOR and MAAP predictions for which the melt is not coolable because only bulk cooling on the top is considered.

USTUTT applied also the MEWA code to the coolability analysis of a melt layer under bottom-injection condition. The experimental database is scarce, and most experiments provide only few measured quantities for comparison with models. Nevertheless, the calculations reproduce major processes observed in experiments, notably the strong initial evaporation, intensive mixing, fast cool-down and freezing of porosities, and the model correctly captures influences of major parameters. The present experimental and theoretical results support the technical applicability of the COMET concept. Further analyses with the model are intended to address specifically the discussion points about freezing at the inlet and crust formation at the top of the melt layer and their consequences. As an example, Figure 11 presents the improved steam flow calculation by MEWA of a previous VULCANO experiment with a COMET core catcher.

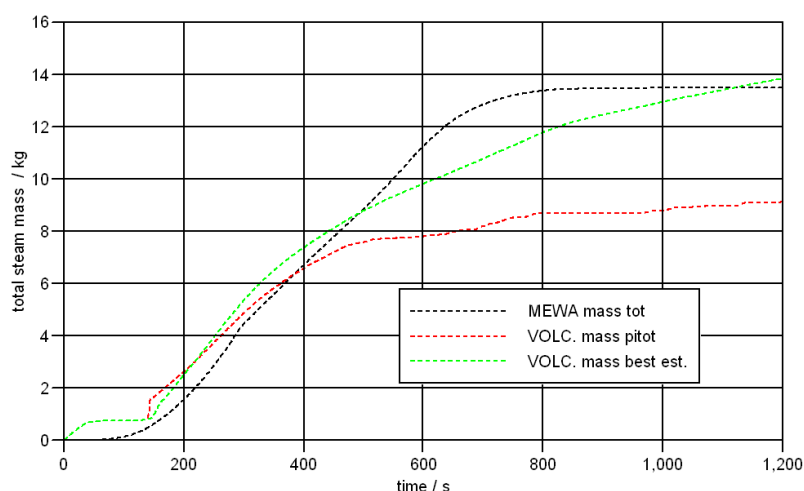


Figure 11: Comparison of (MEWA) calculated and measured total steam mass for the VULCANO VW-U1 experiment with 2 tubes of the COMET PCA concept

These models have been validated on experiments such as VULCANO COMET and VW-U1 (performed by CEA within PLINIUS EU project). They can then be applied to reactor cases. IRSN performed a broad literature review on bottom quenching, about the whole available set of experimental programs (COMET, DECOBI, VULCANO VW-U1, SSWICS12 & 13, INERI and some separate-effect experiments), as well as modelling and simulations.

Potentialities of MC3D for bottom injection calculations have also been assessed by IRSN.

WP6.4 Bringing research results into Reactor Applications (MCCI-RA)

Reactor applications are a necessary step to ensure that the R&D keeps linked to the industrial goals. Discussions have been conducted at the MCCI review meetings on reactor scale calculations made by various partners. In order to have even more fruitful exchanges, a benchmark exercise was performed under the coordination of INRNE. A Station Black-Out scenario (with failure of secondary side BRU-A valve and pressuriser valve stuck open after reaching its set point) for a VVER-1000/V-320 reactor and a mainly siliceous concrete in the cavity has been selected. Seven partners joined the benchmark with 4 different codes. The comparison of the calculation results show that there are no major differences between participant results, at least for the first 50 000 s. The major differences are due to difference in stratification hypotheses.

TRACTEBEL performed ASTEC MCCI calculations on a Belgian PWR on the influence of concrete water content on stratification. It showed a cliff-edge effect between 5.5 and 7.0 wt% of gas content in concrete through the impact on gas superficial velocity and on the metal inventory, leading to

suppression of pool stratification and delayed basemat melt-through beyond some level of concrete gas content.

BARC performed ASTEC (MEDICIS module) calculations of a SA sequence in a PHWR leading to corium interaction with vault concrete.

1.3.4 WP7: Containment

Concerning activities on containment issues, the following benchmarks on available experiments on hydrogen mixing and combustion in containment have been completed: containment spray modelling (on IRSN heat and mass transfer and CALIST experiments), condensation modelling in Computational Fluid Dynamics (CFD) codes (on CONAN experiments at the University of Pisa), hydrogen combustion (on ENACCEF experiments at IRSN-CNRS/Orléans), benchmark on a generic containment model, based on a German PWR, and benchmark on Passive Autocatalytic Recombiner modelling, using experiments from the THAI (Becker Technologies) experimental facility. Hydrogen combustion experiments at Direct Containment Heating conditions were performed in two facilities of different scale at KIT. In the field of ex-vessel fuel-coolant interaction, in complement to the SERENA2 OECD project, analytical and experimental activities continued in the topics of fuel-coolant premixing, steam explosion triggering and explosion occurrence. The MC3D (IRSN), JEMI and IDemo (USTUTT) codes were further developed. Experiments were performed in the KROTOS (CEA), TROI (KAERI), MISTEE (KTH) and DISCO (KIT) facilities.

WP7.1 Ex-Vessel Fuel-Coolant Interaction (FCI)

The SARNET2 project was held concurrently with the OECD program SERENA-2 which was the major frame of international collaboration in FCI, with 12 integral experiments in the TROI (KAERI) and KROTOS (CEA) facilities, accompanied by analytical activities (pre/post-calculations, benchmarks). It was then decided to have in the WP7.1 a complementary work based on discussions and comparisons of models and code behaviour. National experimental activities were also discussed and an experiment was proposed in the DISCO (KIT) facility in the frame of the LACOMEFCO FP7 project. The present conclusions are referring to the global improvements performed during the SARNET2 period, with emphasis of the specific WP7.1 activities (for more details, see the Deliverable SARNET2-CONT-D7.5).

It is first to be recalled that the analytical preliminary phase of the SERENA project (SERENA-1) was concluded with the recognition that an explosive interaction should not bring a failure of the vessel in most cases. Such favourable conclusions could not be drawn for the ex-vessel situation, due to the weaker strength of concrete structures. Although R&D activities are pertinent to both situations, the applications are done with reference to the ex-vessel case.

Both SERENA-2 and SARNET2 main orientations of work for improvements were: melt fragmentation, melt solidification, and void production and impact on steam explosion, with applications as close as possible to the reactor case (material, geometry).

Regarding melt fragmentation, the analyses and discussions in the WP7.1 addressed the consistency of the MC3D and JEMI models, both codes being those who had the major improvement worldwide in the considered period. The outcomes of this activity were nevertheless developed in the conclusion document of the SERENA-2 project.

An outcome of the WP7.1 analytical work is that simple models based on the Kelvin-Helmholtz (KH) instability might be preferable to complex models as those foreseen previously. It was concluded that the KH model leads to a physically consistent basis for modelling. The problems related to drop size (generally considered as too small) can be overcome first by a physically based analysis (the KH model is consistent with the classical Weber criterion) and second by a more

precise application of the model (taking into account the entrainment of the ambient fluid with the melt).

It is nevertheless important to point out current difficulties in modelling:

- Despite its apparent simplicity, the introduction of the KH model in a CFD code is difficult when it is based on the local properties (which is nevertheless important for the capabilities of extrapolations to situations not covered by experiments) due to the difficulty of properly evaluating these properties (and avoiding mesh sensitivity).
- The fragmentation occurs in a multiphase unsteady environment whereas KH considers a monophasic steady ambient fluid. The modelling for such conditions should receive a particular attention.

It is also important to highlight that this work triggered discussions regarding the mean corium drop diameter to consider for comparison of the calculations with the experiments (MC3D using a mean Sauter diameter whereas JEMI uses a mean mass diameter). It was concluded that the use of a multi-size approach would be helpful.

Melt solidification was (and is still) considered as the major effect limiting (and possibly inhibiting) steam explosions so that an accurate modelling of the phenomenon is of first priority. The actions went in two complementary directions for the MC3D and JEMI codes:

- Models for predicting drop resistance to pressure perturbations;
- Development of multi-size group methods to describe the drop population.

It is to be highlighted that the SERENA-2 project allowed confirming the relatively small (size of the melt drops during the premixing).

Void production and impact on steam explosion was an important topic of work. Void is suspected to be an important effect limiting explosions in the SERENA-2 KROTOS experiments. And the X-ray images confirmed the presence of high and transient local void. At reactor scale, most of the calculations tend to predict a quite large void around the melt jet. This can be explained simply by geometrical effect related to the large scale. It is nevertheless to be admitted that at the moment, the impact of the void is still very uncertain. The impact in the codes was analysed in the SARNET frame by comparisons of calculations with JEMI and MC3D of simple analytical cases which are not necessarily realistic: 1D situations (up to steady behaviour); 2D situations at experimental scale; and 2D situations at reactor scale.

This sensitivity study could reveal the intrinsic behaviour of the codes. The conclusions are as following:

- The loads evaluated by the two codes, although with very different models, are of the same order of magnitude.
- In both cases, the impact of reduction of load is not clear, except for very high void.
- For 2D cases (see Figure 12), despite high pressure, the void never really collapses, except partly and locally at the passage of the shock front. Water is entrained inside the mixture from the outside and rapidly evaporates.
- As the shock proceeds faster at low void, the fragmentation is also incomplete when reaching the top. At high void, the fragmentation is more extensive but is in fact mostly due to fragmentation in the gas (high velocity). Their impact on the explosion strength is not certain. As a consequence, not all the small particles recovered in the experiments might have participated efficiently in the explosion.

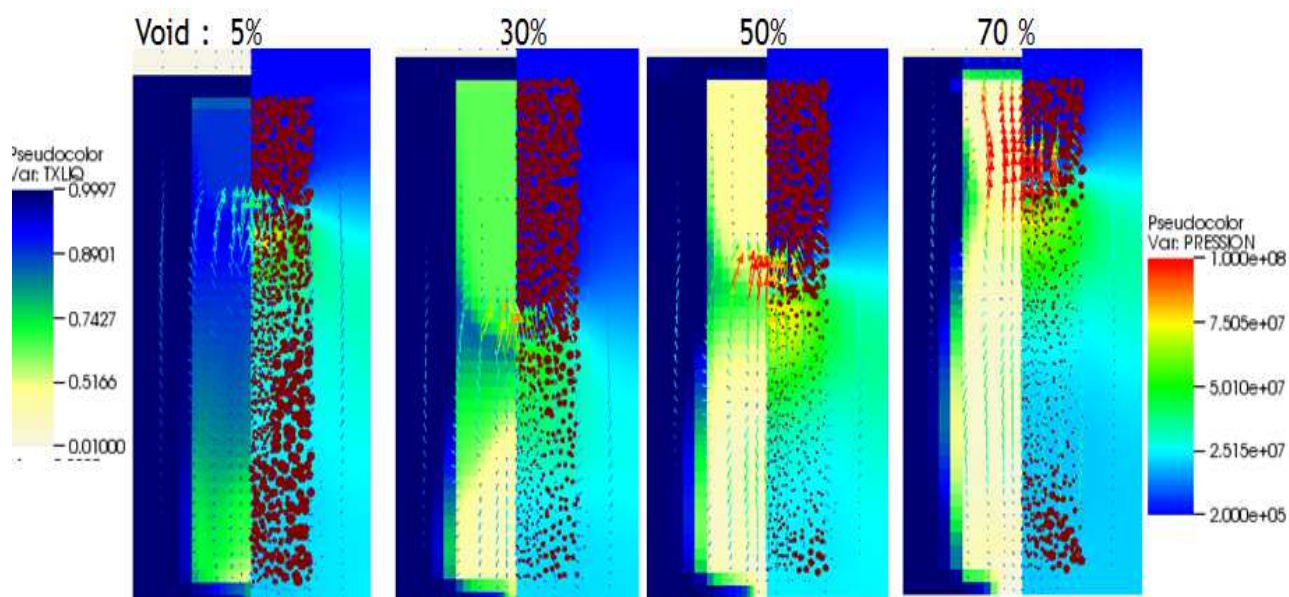


Figure 12: Visualization of the flow in MC3D calculations in the experiment-scale 2D study. Each picture represents a configuration at a given time for 4 different initial void fractions (left part of each picture: amount of liquid (TXLIQ) and liquid velocities (arrows); right mirrored part: pressure in Pa and gas velocities (arrows); the points represent the melt fraction)

On steam explosion triggering

Apart from these investigations, other analyses were conducted, based on experiments. Those cited below confirmed a tendency to spontaneous explosion triggering from various flow configurations.

- KIT and IRSN conducted jointly a series of experiments in the DISCO facility (KIT), with geometry roughly typical of a PWR at reduced scale, involving ex-vessel FCI with melt injection from a slightly pressurized vessel (~10 bars). The melt was a mixture of Fe and Al_2O_3 . One experiment was conducted in the LACOMECE frame. Two experiments led to spontaneous explosions, the strengths of which were moderate due to late triggering of the explosion after the start of the ejection. The calculations performed with MC3D indicate two possible triggering events: a possible flow back of the water on the jet and the change of configuration of melt flow at the break. This indicates that in real reactor situation there are several possibilities of triggering.
- The recent PULiMS experiments at KTH investigated the effects of underwater liquid melt deposition on the pit floor and led to spontaneous powerful explosions in several experiments. This is an important issue because in case of large jets or small water height, the melt should effectively spread as a liquid on the pit floor and this might participate to the explosion, or even trigger it. Currently, the codes capabilities related to this effect need to be largely improved.

WP7.2 Hydrogen mixing and combustion in containment (H2)

In this sub-WP, phenomena linked to the hydrogen-in-containment issue were addressed. Containment thermal-hydraulics, including hydrogen distribution, different hydrogen combustion regimes, their impact on containment structures and measures to prevent combustion processes or at least to mitigate their consequences were covered. The work was divided into 4 separate tasks.

Modelling of containment sprays

The objective of this task was to perform code benchmark exercises (organised by IRSN) on analytical experiments concerning phenomena involved in containment depressurisation and mixing by spray systems. Code-experiment as well as code-to-code comparisons were performed.

The first phase of the benchmark consisted in simulating the “elementary” heat and mass transfer CARADAIS experiment (IRSN): separate uniform droplets were generated, dropped and observed in a vessel with a controlled atmosphere. The simulations enabled the assessment of relevant heat and transfer models in computer codes. The following organisations participated in the benchmark: EDF, GRS, IRSN, KIT, LEI, NRG, UJD SR, UJV and UNIP. The main result is that the mass transfer modelling is responsible for most of the observed differences. Each term involved in the modelling can lead to differences in the results that can be enhanced as soon as the droplet and gas temperatures change significantly.

The second phase of the benchmark consisted in simulating an experiment in the CALIST (IRSN) facility which provided data on momentum transfer on a single PWR spray. The following organizations participated in the benchmark with mainly CFD codes: EDF, IRSN, KIT, NRG, RSE, UJV and VTT. Measured and calculated droplet and gas velocities in different regions of the spray were compared. It was found that droplet velocities were well reproduced at a location 1 m below the spray nozzle, even if the spatial spread of the spray was not reproduced. Concerning gas velocities, most of the codes found values of the vertical velocities between 1 and 2 m/s around the spray, indicating large differences between the participants on the gas “external” entrainment (i.e. out of spray zone). For the internal entrainment (i.e. within the spray zone), the only experimental data available are the gas velocity profile at the nozzle outlet. None of the codes was able to reproduce the shape of the spray and thus the initial entrainment rate (Figure 13). This result revealed an open issue on which research is still necessary.

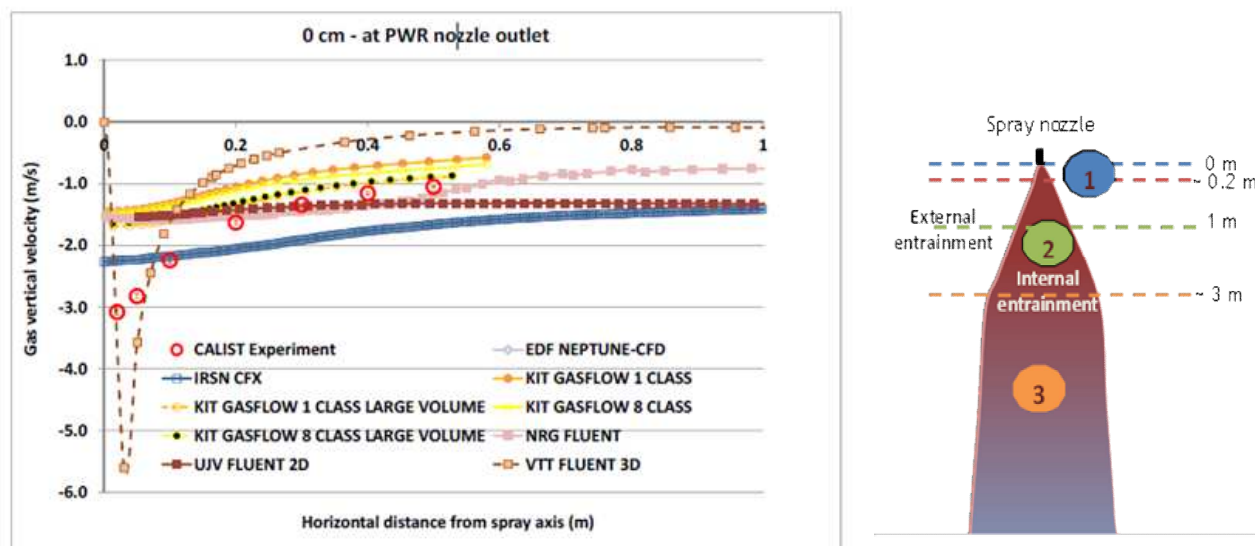


Figure 13: Benchmark on CALIST spray experiments – Left: profile of gas vertical velocity at the spray nozzle exit (blue line on right scheme) – Right: different zones of the spray impact

Condensation modelling in CFD codes

During a SA, steam condensation would occur on containment structures. The modelling of condensation is thus important for the prediction of containment thermal-hydraulics.

A benchmark exercise on steam condensation, organized by UNIP, was completed in two phases. Experiments on steam condensation that were performed in the CONAN facility (UNIP) were

simulated. The following organisations took part in the benchmark (either in both or a single phase): CEA, FZJ, GRS, JSI, KIT, NRG, NUBIKI, UJV, UNIPi and AECL.

Concerning the first phase of the benchmark, the condensation rate in different experimental runs was well predicted. In particular, detailed models (where diffusion was explicitly modelled, with a mesh refinement close to the wall) were able to provide very good results. More approximated modelling techniques also provided acceptable results. This confirmed the adequacy of current models for the considered physical conditions.

In the second phase of the benchmark, the experiments were performed with a much lower gas mixture velocity, which is closer to the natural convection conditions which are expected inside the containment during a SA. In general, most codes provided adequate results: the lesson from the previous phase seems to have been learned (Figure 14). In particular, the models based on diffusion layer theories caught the main features of the condensation rate trends. Interesting applications of analogy-based models seemed also sufficiently successful for being trusted in practical applications. Finally, the results showed that simple formulas for heat and mass transfer coefficient seem not to be trustworthy for general values of steam concentration decrease.

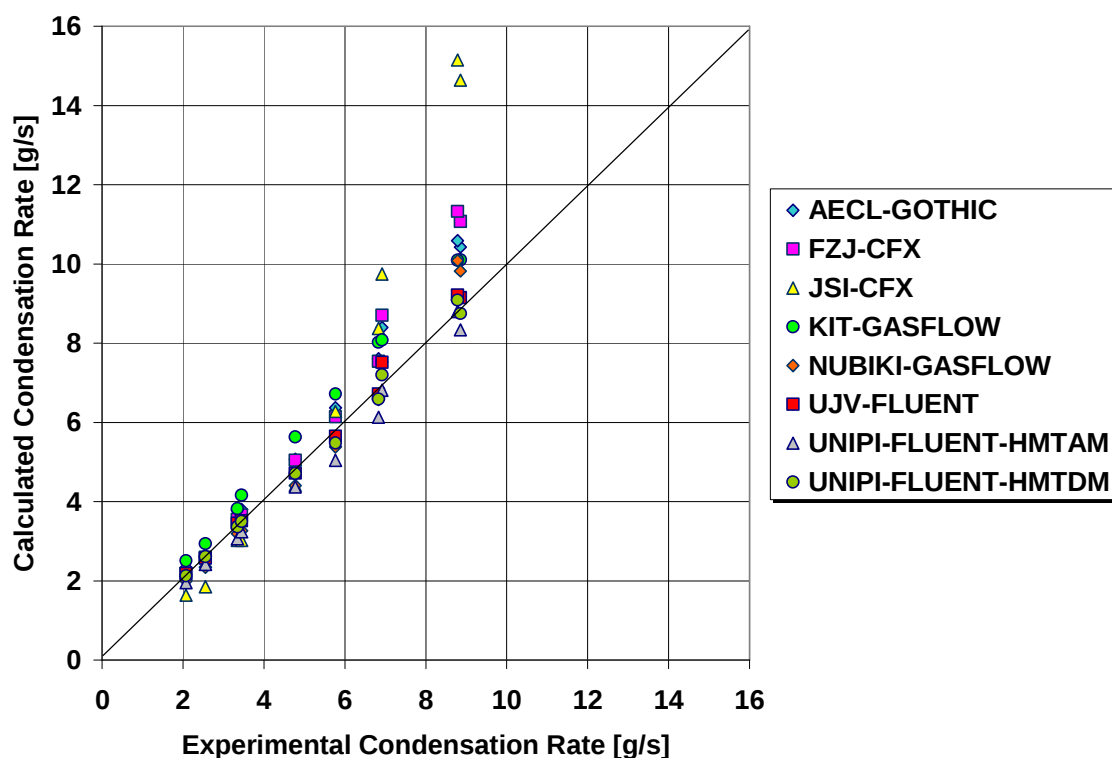


Figure 14: Benchmark on CONAN steam condensation experiments – Evolution of condensation rate

Benchmark for flame acceleration and deflagration-to-detonation transition

In case of a SA, hydrogen could be generated and released in the containment, creating a combustion hazard. The planned work aimed at a better understanding of the phenomena of flame acceleration and deflagration-to-detonation transition.

A benchmark exercise on hydrogen combustion, organized by IRSN, was completed in two phases. The benchmark consisted in simulating experiments, performed for IRSN in the ENACCEF experimental facility (Figure 16), located at the ICARE institute of the CNRS research centre in Orléans (France).

In the first phase of the benchmark, the simulated tests dealt with the effect of turbulence on flame propagation in hydrogen/air mixtures. Experiments were performed with different blockages of the acceleration tube. The following organisations participated in the benchmark: AREVA, GRS, IRSN, LEI, NUBIKI, RUB and KAERI. The results showed that codes are able to predict the pressure maximal value and the combustion completeness. Codes also describe the effect of turbulence on flame acceleration and deceleration. Nevertheless, improved models are needed to well describe the turbulence-flame interaction for the quasi-laminar flame regime and for acceleration and deceleration phases. Also, heat losses are still underestimated by all codes.

In the second phase of the benchmark, the simulated tests dealt with the effect of diluents (mixture of He and CO₂, which acted as a substitute for steam) on flame propagation in hydrogen/air mixtures. Experiments were performed with different diluent concentrations. The following organizations participated in the benchmark: GRS, IRSN, NUBIKI, KAERI, JSI and NRG. The results of the blind step showed that both CFD and lumped-parameter codes are able to predict acceleration and deceleration phases, as well as the pressure maximum value (Figure 15). The open phase will be completed after the end of the SARNET2 project.

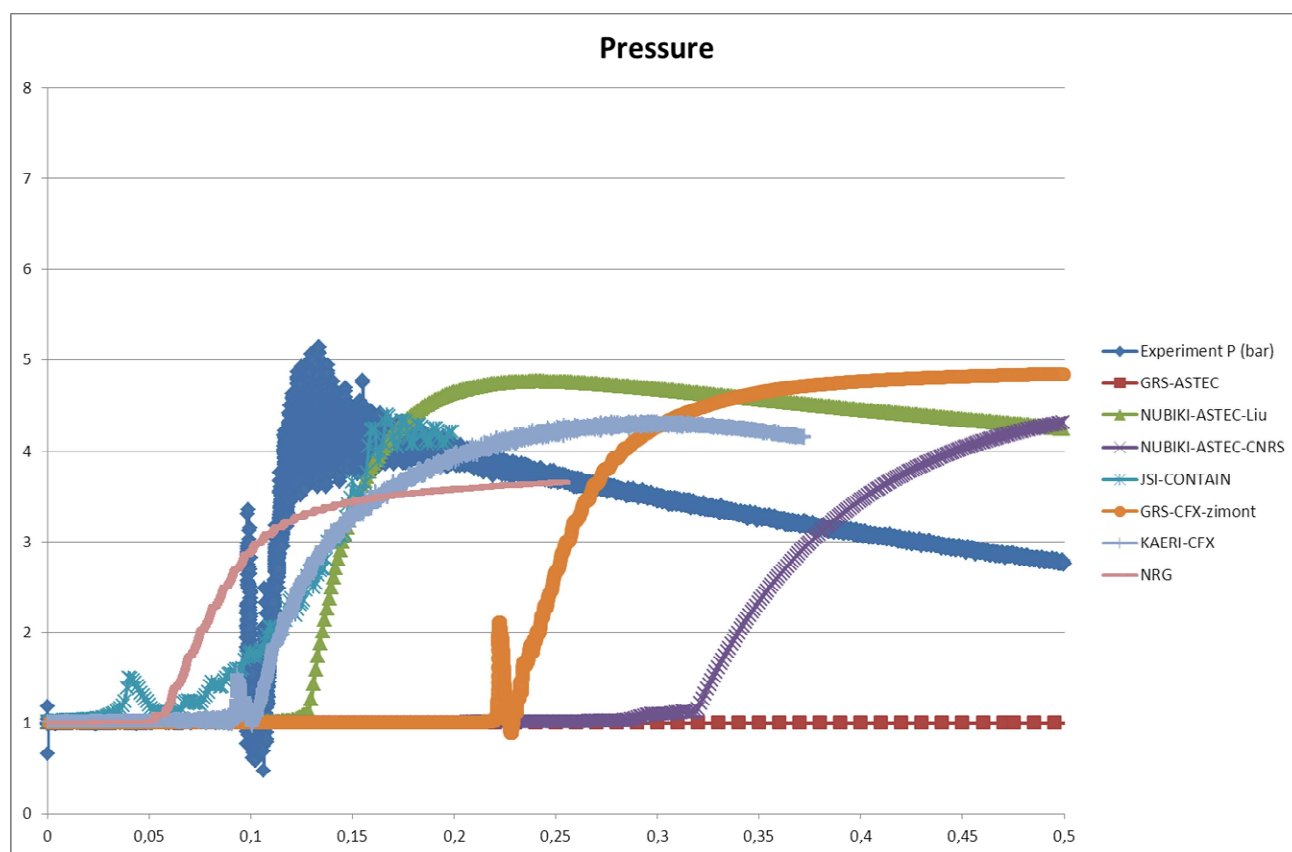


Figure 15: Blind calculations of pressure evolution in the benchmark on ENACEFF combustion experiments

Hydrogen combustion in the course of DCH

Hydrogen combustion can be the dominant contribution to containment pressurization during a Direct Containment Heating process. Experiments in two KIT experimental facilities have been carried out: 8 experiments in the small-scale DISCO facility (volume 13,9 m³) and 9 experiments in the large-scale A2 facility of HYKA platform (volume 221 m³).

Simulations of experiments performed within the preparation phase have led to the preliminary conclusion that three regimes of combustion can be distinguished:

1. At low initial hydrogen concentration, hydrogen injection leads to formation of attached diffusion flame.
2. If hydrogen concentration is slightly below the lower flammability limit, injection of small amounts of hydrogen can lead to fast formation of large-scale burnable mixtures.
3. If hydrogen concentration is higher than the lower flammability limit, ignition of the burnable cloud results in different modes of premixed combustion. The flame speed and connected pressure growth can be different depending on turbulence level, obstruction of the volume, etc.

Hydrogen combustion tests at DCH conditions conducted in the two above-mentioned different size facilities have shown that there is no scaling effect relative to the pressure increase in the containment. The pressure increase correlates with total hydrogen burned. The fraction of hydrogen that burns depends on the ratio of pre-existing to blow-down hydrogen and on the total amount of hydrogen and varies between 46% and 100%. Compartments may have an effect on the burnt fraction but this has not been investigated in depth. The efficiency of combustion energy conversion into pressure varies between 42 and 71% and again may be affected by compartments and structures in the containment. These effects can be analysed by code calculations, for which the experimental results may serve as a data base for model improvement and validation.

WP7.3 Bringing research results into reactor application (CONT-RA)

The work in this sub-WP was divided in 4 separate tasks.

Ex-vessel fuel-coolant interaction reactor application

The main activities in this task have been presented above since in close link with WP7.1:

- Comparison of calculations with JEMI and MC3D codes on simple analytical cases in 2D situations at reactor scale,
- Interpretation with MC3D of the DISCO experiments with geometry roughly typical of a PWR at reduced scale, involving ex-vessel FCI with melt injection from a slightly pressurized vessel, which indicated several possibilities of steam explosion triggering in real reactor situations.

Detailed evaluation of important experiments

Within the topic of atmosphere mixing, some experiments that have already been simulated in the past were considered again, with additional knowledge and experience gained from work performed in the meantime. The performed simulations confirmed that older experiments are useful for the assessment of newly developed codes or new modelling techniques with existing codes, and should be used extensively for that purpose.

An experiment on atmosphere mixing performed in the THAI experimental facility (located at Becker Technologies, Germany) was simulated with the COCOSYS computer code by RUB. A special nodalisation was used to successfully simulate a rising steam plume.

JSI simulated with the ASTEC and CONTAIN codes an experiment on non-homogeneous atmosphere, performed in the TOSQAN facility (IRSN) and used for the OECD International Standard Problem N°47. The vessel was subdivided in such a way as to reproduce natural convection loops, observed in the experiment.

An experiment on atmosphere stratification, performed in the MISTRA facility (CEA), was simulated by LEI with the COCOSYS code. The simulation confirmed the necessity of an appropriate nodalisation to simulate the stratification breakup with a steam plume.

Preparation of generic containment input deck and test application

The main purpose of this task was the development of a “generic containment model”, which will be suitable, first for comparisons of different codes, and later for considering the issue of scaling between experimental facilities and actual containments.

A benchmark exercise was organized by Jülich in three phases. The purpose of the benchmark was to compare the results of lumped-parameter codes by simulating a transient in an “idealized” containment model, based on the containment of a German PWR. The “idealization” of the containment model does not mean that the model was not realistic, but that all its characteristics were precisely defined. In this way, the comparison of results obtained with different codes should not be distorted by different modelling of the same actual containment due to different interpretations by code users. The following organisations participated in the benchmark (at least in one of the phases): AREVA, ENEA, GRS, IRSN, JSI, Jülich, NRG, NUBIKI, RSE, RUB, UJV, UNIPI, VTT and VUJE.

The results of the first phase, which involved the simulation of a loss-of-coolant accident, were compared among each other as well as against a common solution, i.e. an average with tolerance band. The comparisons revealed some deficiencies in the input decks and models, and motivated sensitivity analysis on e.g. nodalisation of the heat conducting structures, the time step management or the treatment of source terms. The main outcome of the first phase was the confirmation of the significant influence of so-called “user effects”.

For the second phase, the same containment model was used, with the addition of hydrogen injection in the SA scenario. Detailed comparison and analysis were performed for the following simulated physical variables, phenomena and models: flow pattern between different control volumes, integral balances of mass and energy, sump heat and mass transfer, droplet treatment, thermal radiation, treatment of steam and water injection during blowdown, characteristic length scales for heat transfer simulation and structure nodalisation. The main outcome of the second phase was the evidence of the influence of different physical modelling between the lumped-parameter codes (Figure 16).

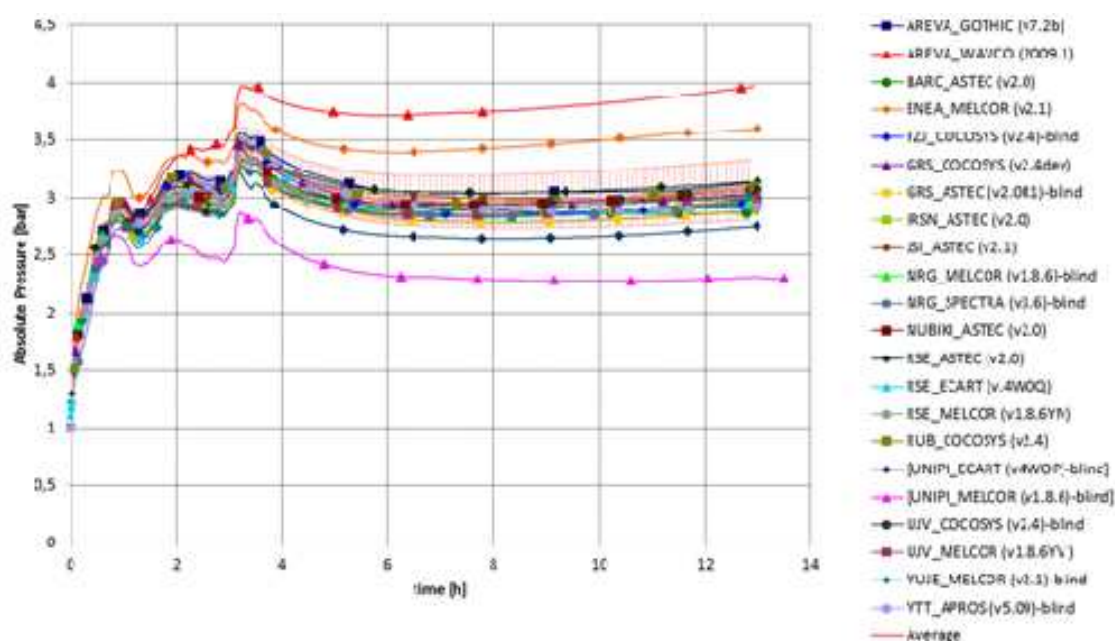


Figure 16: Benchmark on a generic containment model – Evolution of total pressure

The third phase consisted in the simulation of the same transient as in the second phase, with the addition of Passive Autocatalytic Recombiners (PARs) in the containment. The comparison and analysis of results revealed the differences in the applied PAR models. In general, the inclusion of

PARs led to lower hydrogen concentrations and slightly higher gas temperatures and pressure. The main outcome of the third phase was the actual application of the developed generic containment model to a comparison of specific PAR models.

Analysis of PAR modelling

The interaction of PARs with the containment atmosphere is one of the most important issues of interaction between the atmosphere and hydrogen mitigation systems.

A benchmark exercise was organised by GRS to investigate different modelling of PAR devices. Two relatively simple experiments using PARs were simulated (both atmosphere thermal-hydraulics and recombination). Both experiments were performed in the THAI containment facility by Becker Technologies under dry air conditions and different initial pressures. The following organisations participated in the benchmark: GRS, IRSN, LEI, NRG and NUBIKI.

The benchmark provided information how well the different codes simulate the following: pressure increase and decrease, thermal stratification and recombination rate. At present time, CFD codes appear to be more adequate to model the thermal stratification than lumped-parameter codes.

1.3.5 WP8: Source term

The WP8 goal was to reduce the uncertainties associated with calculating the potential release of radiotoxic fission products, concentrating on iodine and ruthenium. The specific objectives were as follows:

- Oxidizing impact on source term: fission products release from fuel, including high burn-up fuel; ruthenium transport in RCS and behaviour in containment.
- Iodine chemistry in RCS and in containment.
- Bringing research results into reactor application: benchmarking of available codes against integral experiments; maintenance of data books where existing, such as for iodine, and development, such as for ruthenium.

WP8.1 Oxidising impact on Source Term (OXI)

Fission products release from fuel

AECL and CEA provided experimental data on fission product release from fuel under oxidising conditions.

In the frame of the International Source Term Programme (ISTP), the VERDON-1 test was successfully performed by CEA on September 30th 2011. It was devoted to characterize the fission products release from a high burn-up UO₂ fuel under reducing conditions at very high temperature (up to 2600°C). The VERDON-2 test was performed by CEA on June 27, 2012: it was dedicated to study the impact of air ingress on fission products release and transport. The fuel sample was a MOX fuel irradiated at 60 GWd/t. VERDON-3 and -4 tests are two complementary tests devoted to characterize the fission products release from the same MOX fuel used in VERDON-2 under respectively oxidising (VERDON-3) and reducing (VERDON-4) conditions for the last high temperature phase. The VERDON-3 test has been performed after SARNET2 timeframe on April 17th 2013. VERDON-4 is planned to be performed in the first semester of 2014. The VERDON Working Group started its activities in January 2012. The aim of this group was to simulate the VERDON experiments with different computer codes (SOURCE, MAAP, ASTEC, ATHLET-CD and MELCOR) and to compare the codes and the simulation results. The activity of VERDON Working Group will continue after SARNET2 as data from the remaining tests will become available.

AECL provided data from their MCE1 and HCE3 fission products release experiments to IRSN and EDF for model validation.

INR continued to study self-disintegration of sintered UO₂ pellets in air-steam atmosphere using the modified FIPRED EQ equipment.

In the framework of the QUENCH program of KIT, experimental investigations of the air oxidation behaviour of Zircaloy-4 have been extended to the advanced cladding materials M5™, produced by AREVA, and ZIRLO™, produced by Westinghouse, with focus on the reaction of oxygen-stabilised¹ α -Zr(O) phase with nitrogen.

As to theoretical tasks, IRSN and ENEA conducted the development and assessment work of some fission products (Ru, Mo...) release models in ASTEC (ELSA module). EDF improved models for its proprietary version of MAAP and validated them on nine VERCORS tests, including RT8 under pure air. GRS extended fission products release modelling in ATHLET-CD module FIPREM, so that the calculated release is a function of partial pressure and depends on the temperature, the pressure of the system, and some material properties. PSI implemented a new air oxidation model in a special version of MELCOR 1.8.6 and a developmental version of RELAP5/SCDAPSim. PSI performed analyses of air ingress experiments PARAMETER SF4 (in Podolsk, Russia), QUENCH-10 and -16 as well as OECD SFP. USNRC modelled fission product release from high burn-up uranium oxide and mixed oxide fuels.

Ruthenium behaviour in containment

In the frame of analysis of ruthenium behaviour in the containment, analysis was based on EPICUR tests carried out by IRSN. These tests were conducted as a part of the ISTP.

Fission product transport in RCS

AEKI carried out the RUSSET-8 program, in which the influence of different surfaces (stainless steel, E110 cladding material, alumina) on the transport of ruthenium oxides was studied in the temperature gradient. JRC-ITU performed a series of 8 revaporisation tests using the deposits from the vertical line above the Phébus FPT3 test bundle. A final report on the experiments was delivered to IRSN which included it in the FPT3 final report.

IRSN and ENEA have developed and assessed ruthenium transport models in ASTEC V2.0 using data from AEKI and VTT experiments. VTT shared results from CFD modelling of the ruthenium transport experiments. USNRC has prepared a data base on the gas phase Ru-O-H system thermochemistry.

Conclusions from WP8.1

Ruthenium release from the core is known to depend on state of the cladding material. As a result of the numerous separate-effect tests that were carried out in SARNET2, a model for fuel cladding oxidation in air containing atmosphere could be formulated.

AEKI, ENEA, IRSN and VTT have prepared a joint synthesis paper “Transport of ruthenium in primary circuit conditions during a severe NPP accident” for ERMSAR 2013 conference. AECL, ENEA and IRSN are preparing a joint synthesis paper “Validation of Ruthenium Release Models using AECL Fission Product Release Tests”.

Phébus FP experiments indicated that the ruthenium deposits on the fuel bundle can act as a source of volatile ruthenium. In the separate-effect tests, ruthenium was transported through the circuit

¹ Diffusion of oxygen through the ZrO₂ oxide layer into the underlying Zr metal during oxidation can prevent the transformation of the low temperature α -phase (for Zircaloy in the range 820-980°C) into the high temperature β -phase up to its melting point (~1800°C), hence the term “oxygen-stabilised α -Zr(O)”.

mainly as aerosol particles when the ruthenium source was oxidized above 1227°C. When oxidation took place at a rather moderate temperature (1000 to 1100°C) a significant fraction of ruthenium was transported as gaseous RuO₄. The decomposition process of RuO₄ to RuO₂ was not complete and their vapour pressures did not follow equilibrium concentrations.

The interpretation of experimental and simulation results revealed that the ruthenium chemistry modelling in ASTEC code (SOPHAEROS module) has to be improved especially in the case of air-ingress scenarios, for which the ruthenium releases from degraded fuel could be important. Therefore, thermodynamic data of RuO₂(g) have to be determined and the material databank of SOPHAEROS updated. To assess the possible influence of kinetic limitations resulting from high thermal gradients, the rate constants of RuO₃ oxidation reaction by O₂ or steam should be determined at least. By assuming that the formation of gaseous RuO₄ was kinetically limited in a high temperature gradient, a reasonable agreement was found e.g. between ASTEC modelling and RUSET experiments (Figure 17).

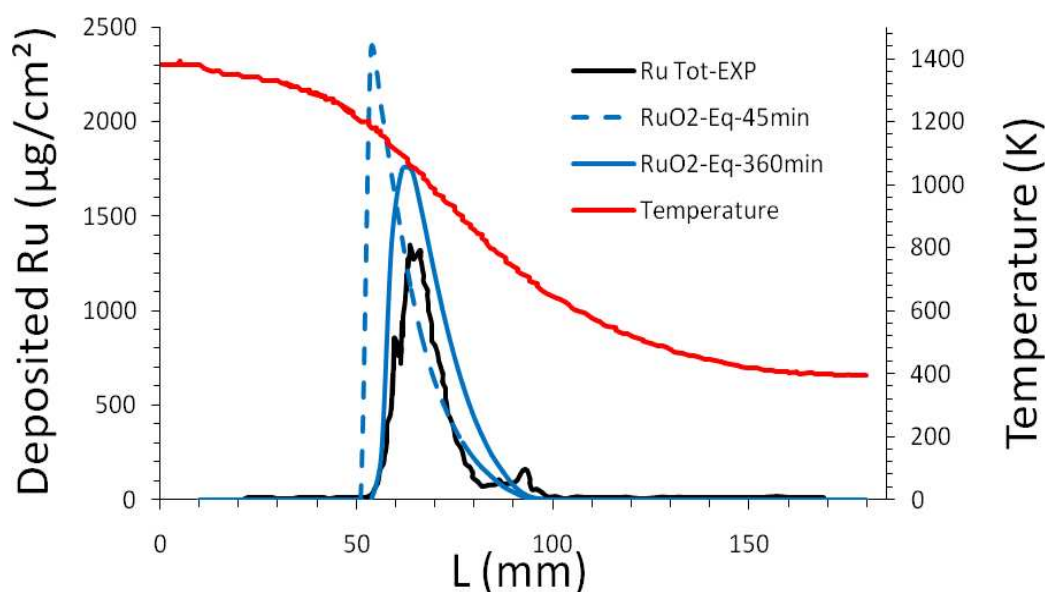


Figure 17: Comparison of the amount of Ruthenium deposited in RUSET test with ASTEC (SOPHAEROS) calculation when assuming that the formation of gaseous RuO₄ was kinetically limited [Kärkelä T. et al. Transport of ruthenium in primary circuit conditions during a severe NPP accident, ERMSAR-2013, Avignon, France, 2-4. Oct. 2013]

For this reason IRSN has launched in the frame of the OECD STEM programme a specific project called START (Study of the TrAnsport of RuThenium in the primary circuit), which is focused especially on air-ingress scenarios. The release and transport of fission products, e.g. ruthenium, from irradiated, high burn-up UO₂ and mixed oxide (MOX) fuels is currently being studied also in the VERDON experimental program led by CEA (see the abovementioned VERDON-1, -2 and -3 tests). A future VERDON-5 test has been proposed in air-ingress conditions using high burn-up UO₂. The data obtained in these and previous programs will constitute a support to the model developments and also be very useful in the validation effort.

WP8.2: Iodine chemistry in the RCS and containment (IOD)

Iodine chemistry in the RCS and containment (BDBA)

IRSN performed 4 tests in the CHIP V1 loop to study the system (Cs, I, O, H) at about 500-600°C. Then IRSN conducted iodine chemistry experiments in primary circuit conditions using the small-scale GAEC and CHIPINIO facilities as well as the second version of the CHIP PL line, with injection of I, Cs and B.

VTT completed 13 experiments on iodine chemistry on the primary circuit surfaces using EXSI-PC facility. A significant fraction of iodine was released and transported as gas at low temperatures. VTT carried out experiments on the radiolytic oxidation of iodine using beta radiation. In addition, the studies on the desorption of iodine from IO_x and CsI deposits on stainless steel, copper, aluminium, zinc and paint surfaces were carried out in collaboration with Chalmers.

IRSN developed a kinetic modelling of the gaseous reactions linked to simplified systems such as {Cs, I, O, H} on the basis of the primary circuit experiments performed. ENEA carried out modelling of inter-granular behaviour of iodine species from AECL GBI experiments GBI4, GBI5 and GBI6.

IRSN contributed through data from 14 EPICUR tests carried out in the ISTP campaign to validate the model of organic iodide (RI) formation from an irradiated painted coupon. The coupons were loaded with iodine and placed in the gaseous phase at various temperature, relative humidity and initial iodine loading. After the ISTP program, EPICUR tests were continuing under OECD/STEM. Data from 14 EPICUR tests were used to validate the model of organic iodide (RI) formation. In addition to iodine-paint interaction, two EPICUR tests studied the formation and stability of the IO_x particles under radiation and one exploratory EPICUR test was performed to study the CsI stability under radiation. Experimental determination of the kinetics of destruction of organic iodides CH₃I, CH₂I₂, C₂H₅I, C₃H₇I by air radiolysis products was also completed at IRSN.

AECL distributed data from five older RTF experiments to the project partners. AECL completed the OECD BIP project and issued a final summary report. Similar work has been started in the immediately following BIP2 project.

The Phébus.FP synthesis report was completed by IRSN and JRC, followed by an open seminar organised in Aix-en-Provence in June 2012 on the experimental results, modelling and applications of the data, the proceedings of which have since been published in a special issue of Annals of Nuclear Energy. In particular, the general conclusions drew attention to how the improved understanding of what are the dominant phenomena governing gaseous iodine behaviour in the circuit and containment re-oriented the R&D programmes, both on experimental and modelling aspects. Although work is still ongoing, the situation is much better than it was before Phébus.FP. Many of the applications by regulators, TSOs... involved improvements to source term evaluation. The GRS team has carried out COCOSYS/AIM code validation particularly on the basis of Phébus FPT2 and FPT1, focussing on uncertain iodine model parameters.

NNL analysed PSI as well as EPICUR data on iodine volatility under sump conditions with INSPECT. NNL performed calculations using the mechanistic IODAIR model to simulate the EXSI-CONT tests.

IRSN pursued a common interpretation of BIP, EPICUR and Phébus FP tests. Modelling of iodine interaction with paints included NRC/SANDIA/IRSN cooperation. Regarding paint-iodine interaction, likely mechanisms of paint ageing that occur during service and during accident conditions were identified as well as the likely mechanisms of iodine interaction with paint. Theoretical studies were performed at IRSN in support of the OECD/BIP2 and OECD/STEM experiments, with recommendations for new experiments in both programmes. IRSN and CIEMAT have carried out analysis of EPICUR experimental results dedicated to organic iodine formation with ASTEC (IODE module). EDF has updated the iodine chemistry model implemented in the MAAP code: this updated model has then been validated on almost twenty EPICUR tests. NNL has made a comparison between the results of EPICUR tests and other similar programmes including the OECD BIP tests. In order to investigate the remaining uncertainties IRSN and USNRC conducted a literature review on mechanistic interaction between iodine and epoxy paints. CIEMAT analysed BIP-RTF tests as well as RTF tests with ASTEC/IODE. USNRC worked on interaction of water with epoxy paints.

Chalmers completed preliminary tests on absorption of I_2 on fresh paint, which is compared with the real paint from Barsebäck NPP. VTT conducted experiments on deposition of iodine oxide particles on painted concrete samples and subsequent desorption of iodine under gamma radiation.

VTT and Chalmers carried out EXSI-CONT tests on radiolytic oxidation of gaseous I_2 and CH_3I in containment conditions. VTT derived a model for reaction between elemental iodine and ozone based on these experiments. Chalmers studied the interactions of IOx aerosols with epoxy paint films and reactive metal surfaces in collaboration with VTT.

The data from THAI program tests Iod-13 and Iod-14, designed specifically to study radiolytic oxidation of gaseous I_2 , were uploaded in the ACT.

On sump chemistry and iodine partitioning, iodine chemistry in the containment sump was experimentally studied at PSI and two extensive experimental reports were released on the effect of impurities on iodine volatility from irradiated water solutions followed by open literature papers, one describing the basis for the modelling at PSI (PSIODINE code) and the other describing the data and interpretation using that model. The aim of the study was to quantify the effects of nitrate, nitrite and chloride ions on iodine radiolysis reactions. The analysis of PSI tests was carried out in NNL with the INSPECT code and compared with earlier tests at Harwell. Chalmers developed an experimental set-up on the volatility of iodine at different redox potentials. USNRC carried out work on the thermal aqueous reactions of iodine and on iodide absorption on debris suspended in sumps. Modelling of aqueous molybdate and borate ions was completed.

As well as iodine-related work, revaporisation tests of FPT3 circuit samples were performed by JRC/ITU with associated tests by VTT, while analysis by *ab-initio* methods has been started by IRSN. Results of recombiner tests in the FPT3 containment showed no evidence that recombiners were poisoned by fission products or by carbon monoxide (from B_4C combustion) during their 30 min exposure time, as described in a journal paper that illustrates as well related experimental results and modelling.

Iodine chemistry in the RCS and containment – Design basis accidents (SGTR)

The activity related to iodine spiking events consisted mainly on gathering and analysis of available plant data. EDF provided iodine activities in the primary coolant system in a French 900 MWe NPP based on the Experience Feedback during normal operations and power transients, and also the design values used for the radiological consequences assessment of DBA accidents. NNL provided data on iodine activity in coolant during SGTR sequences from Sizewell B PWR coolant measurements during operation and shutdown. NNL also analysed ^{131}I activity spikes in Sizewell B coolant during cycle 10, finding reasonable consistency. NUBIKI provided ^{131}I spiking data from Paks VVER NPP from 4 different shutdown events, ^{131}I activities are given during 100% power operation before the shutdown, along with the maximum value measured during the spiking phenomenon. TUS provided a summary of methods used to monitor primary circuit iodine in VVER-440, along with the activity of other radionuclides. USNRC made available a document which provided lists of openly available iodine spiking data and models. IRSN performed analysis of the existing experimental data from tests relevant to SGTR events (KWU, Wale and ARTIST phase VI). Based on the existing data on the predominant iodine chemical reactions and on the partition coefficients of the iodine species, a preliminary investigation of the potential iodine releases was performed. It was concluded that iodine spiking issue has lower risk significance than was expected. The risk has decreased with improving fuel cladding materials. There is still interest in model improvement (e.g. for ASTEC) but not on an urgent timescale. A PhD in IRSN on modelling the iodine spiking issue was due to be defended in December 2013, which should lead to improved ASTEC modelling.

Conclusions from WP8.2

Regarding iodine transport within the RCS the work during SARNET2 focused on explaining the higher than expected gaseous iodine formation observed in Phébus FP tests. Separate-effect tests, with associated modelling, e.g. for ASTEC, were carried out with simulant materials as well as with samples acquired from the Phébus FP circuit. The main phenomena studied were the possible kinetic limitations in the gas phase reactions of iodine and the role of surface reactions on the gaseous iodine fraction. It has been observed that a large fraction of even highly volatile fission products are deposited within the RCS in places with high temperature gradients. The thermal-hydraulic conditions, gas composition as well as other deposited fission products and structural materials influence not only the timing but also the speciation of iodine release from the RCS. The separate-effect tests showed that the gaseous iodine fraction may indeed become very high due to reactions with other fission products and structural materials. The release of gaseous iodine especially at a moderate (400°C) temperature has been both significant and quite unexpected. The modelling work of the influence of surface reactions on iodine speciation is still only starting at the moment. Also further experiments on iodine chemistry within the RCS continue after SARNET2.

Traditionally it has been thought that after about 24 hours, the production of volatile iodine species by radiation chemical reactions occurring in water volumes will dominate the airborne iodine inventory in the containment, which is available for release to the environment through containment leakage or breach. Radiolytic oxidation is the main mechanism for oxidation of iodide in solution to volatile forms such as I_2 . The effect of water impurities on iodine volatility was experimentally studied and successfully modelled by SARNET2 partners. I_2 can react with surfaces in the containment, leading to its retention but also to the potential formation of volatile organic forms. Therefore, iodine-paint interaction was extensively studied in several experimental programs.

The experiments on iodine oxidation demonstrated also that gaseous iodine species such as molecular iodine (I_2) and volatile organic iodide (CH_3I) can react with the products of air radiolysis to form iodine oxide particles. It would be expected that radiolytic processes would be sufficient to sustain gas-to-particle conversion in reactor containment atmospheres following an accident transient. With this radiolytic conversion in mind, it was possible to outline a cycle in which the gas phase oxidation of iodine takes part. The cycle, shown in Figure 18, begins with iodide, which is believed to be the species of iodine prevalent at the start of a postulated reactor accident. Iodide is oxidised to I_2 by water radiolysis products, and partitions to the gas phase. In the gas phase, I_2 reacts to form iodine oxides that are non-volatile at containment temperatures. These oxides will form aerosols that will eventually deposit on containment surfaces. It has been shown that these oxides can be decomposed by the action of heat and/or radiation to re-release gaseous iodine species, or yield iodate when dissolved in water. It has also been shown that aqueous iodate is easily converted to other iodine species (mainly iodide) upon irradiation. The re-formation of iodide completes the cycle.

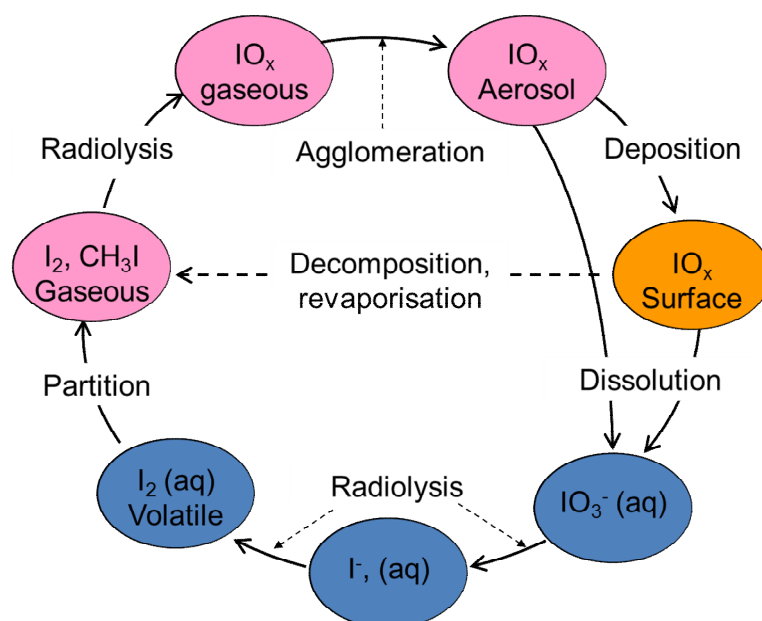


Figure 18: Simplified Reaction Scheme of iodine behaviour within the containment [Dickinson S. et.al. Experimental and modelling studies of iodine oxide formation and aerosol behaviour relevant to nuclear reactor accidents, ERMSAR-2013, Avignon, France, 2-4. Oct. 2013]

The gas-to-particle conversion process could thus maintain the airborne radionuclide contamination of the containment at a higher level than would be obtained from simple equilibration of volatile species between the atmosphere and water volumes or surfaces. Any prediction of the long-term airborne iodine activity levels therefore needs to take full account of all the processes contributing to the steady state. On the other hand, information on iodine oxidation is already applied as the systems for SA source term mitigations are studied within the PASSAM new FP7 project that started in 2013.

WP8.3 Transposition of the R&D results to the reactor scale (ST-RA)

Release of data from THAI Iod-11 and Iod-12 was agreed. USNRC collected additional information on the reactor accident at Three Mile Island. USNRC also made reports on a large number of source term related accident sequence analyses available to the partners.

THAI Iod-11 and Iod-12 Benchmarks

The benchmark exercises were coordinated by GRS. Six other SARNET2 partners (AREVA NP GmbH, CIEMAT, FZ-Jülich, TUS, UNIPI and VTT) participated in the work. The aim was to assess the capability of severe accident codes to simulate iodine transport and behaviour in sub-divided containments. Nine calculations were submitted for Iod-11 and eight calculations for Iod-12. The main findings of the THAI Iod-11 and Iod-12 benchmarks were as follows. As for thermal-hydraulics, some variables like gas temperature and pressure seem to be well captured by codes. However, others like relative humidity (especially) and condensation rates showed a broad scatter. The accuracy of the helium distribution calculated by the codes looked better during steam injection periods. Concerning iodine behaviour, the predictions of gaseous local iodine concentrations were diverse. In Iod-11, for the dome, three calculations showed only minor deviations from the data, as illustrated in Figure 19. However, in the lower compartment, predictions were typically 1-2 orders of magnitude too high. In Iod-12, no code gave a satisfactory evolution of the iodine concentration in all compartments of the vessel. A number of potential areas for improvement have been identified, e.g. I₂-steel interactions and iodine wash-down modelling. As in other similar benchmarks, e.g. OECD/ISP-46 (Phébus FPT1) a large “user effect” was seen, i.e. results obtained with the same code differed considerably, indicating a continued need for thorough user training. The results of the

THAI benchmarks were presented in the ERMSAR 2012 meeting. Also a scientific article on the benchmarks has been published in the Nuclear Engineering and Design journal.

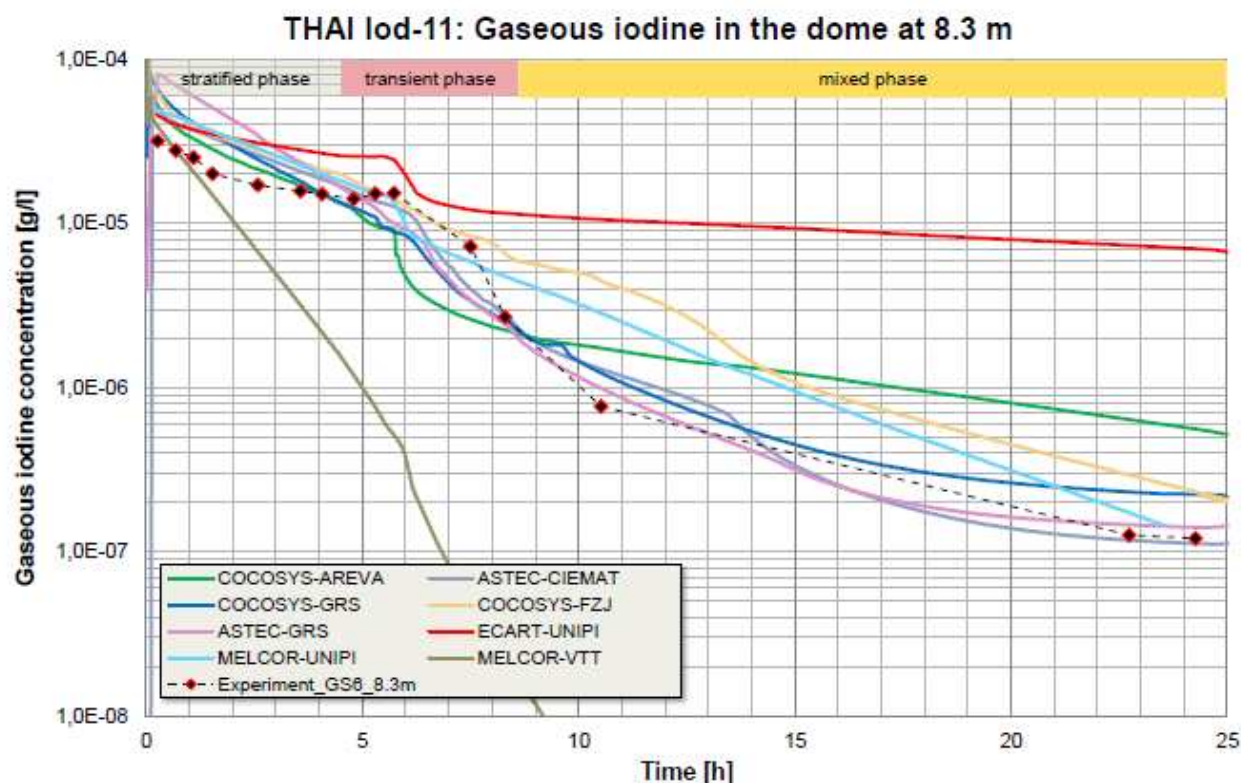


Figure 19: Comparison between the measured and calculated gaseous iodine concentration in the THAI dome part in Iod-11 test [Weber G. et al. SARNET2 WP8 Benchmark on THAI multi-compartment iodine tests – Results for test Iod-11, 5th ERMSAR, Cologne, Germany, 21-23. March 2012]

Phébus FPT3 benchmark

The benchmark on Phébus FPT3 integral experiment was initiated by IRSN. It covers: a) fuel degradation, hydrogen and carbonaceous gas production (from oxidation of the boron carbide control rod), release of fission products, fuel, and structural materials; b) fission product, fuel and structural material transport and deposition in the circuit; c) thermal-hydraulics and aerosol behaviour in the containment; and d) iodine chemistry in the containment. Reference calculations with a prescribed nodding scheme and standard code options were required. The participants could also carry out best-estimate calculations with free, more refined options and nodings, to enable a more detailed interpretation of code capabilities. The calculations by 16 participants (CIEMAT, EDF, ENEA, GRS, IRSN, NNL, NUBIKI, RUB, UNIPi, USNRC, VUJE, TUS, KAERI, RSE, UJV, KINS), involved a varied mix of 8 codes (ASTEC, MELCOR, MELCOR/RAIM, MAAP4, ATHLET-CD, COCOSYS, ECART, INSPECT/IODAIR).

The partners submitted their initial calculations by the end of February 2012. The first comparison workshop was held in 26th of November 2012, at which the partners presented their results and the coordinator gave first impressions. After the workshop, a few of the partners conducted revised calculations. A shortage of resources within IRSN stretched benchmark activities beyond the end of the SARNET2 project, and led to a delay in the detailed analysis of the results. A second workshop to review the draft conclusions of the exercise was held on 1st October 2013 in conjunction with the ERMSAR-2013 meeting. Since then, minor comments from the participants on the draft comparison report have been received, and a final draft taking them into account has been prepared. This

benchmark allowed to assess progress of simulation codes since ISP46 on FPT1 and to confirm the importance of the main on-going research programs.

The report is being issued formally in December 2013. After that, open publication in conference(s) and/or a journal will be considered.

Generic containment scenario

NUBIKI carried out activity on a generic containment scenario together with CIEMAT and RSE. A TMI-2 based database was prepared for a PWR containment providing a nodalisation and information about volumes, structures, flow paths and materials.

Ruthenium data book

The work to create a data book on ruthenium chemistry was started in January 2012 with five partners (AECL, AEKI, CEA, IRSN and VTT) participating in the task. The Ruthenium data book summarizes the experimental programs, the experimental facilities and the main results on the release of Ru from fuel and the transport and behaviour of Ru in the primary circuit and containment conditions. Its first version was finalized at the end of April 2013.

1.4 Potential impact (including the socio-economic impact and the wider societal implications of the project so far) and the main dissemination activities and exploitation of results

1.4.1 Socio-economic impact

As the end-products developed by the network (ASTEC, experimental database) may be used not only for R&D activities but also for industrial applications, many European industry and safety authorities (or technical safety organizations) are contributing to SARNET. In return, the end-products that capitalize the large amount of knowledge acquired in this area contribute to a better prevention and mitigation of SA in existing and future European NPPs of diverse types, and thus to the improvement of their safety.

By fostering collaborative work, the role of Europe as world leader in the SA domain has been strongly consolidated. A first illustration is the request by the US EPRI utilities consortium of a SARNET review of their report on the technical interpretation of the Fukushima accidents. This review was done in a very short time and its technical quality was appreciated by US EPRI. Another illustration is the reinforcement of the role of ASTEC code as reference on since European end-users were mostly using, at SARNET's start, integral computer codes developed in the United States, which results in a strong dependence on the US technology.

SARNET has clearly become a reference for SA research priorities and has already some impact on national programmes. This should be reinforced by the integration in the NUGENIA association. Progressively all the research activities in this field are becoming strongly coordinated by the network, which contributes to an optimised use of European resources.

Through education and training programmes, SARNET has developed synergies with educational institutions, either universities that hosted the educational courses and seconded students in other laboratories, or international entities such as the ENEN network in Europe. This allowed keeping attractive the concerned domain of activity for students and young researchers. The network provides also a wide panel of competences for supporting the emergence of new nuclear countries. All these activities have contributed to enhance and preserve in a sustainable way the European scientific leadership.

Performance indicators had been defined at SARNET2 project' start to assess the progress of the network and the success of integration. The main following conclusions (see the deliverable SARNET2-MANAG-D1.9) are:

- Strong participation of SARNET and non-SARNET organizations to ERMSAR conferences and Education courses, with increase along the project duration. This underlines the improvement of network attractiveness and of involvement on networking activities, especially for young researchers and students,
- Very strong increase of the number of publications in journals or conferences, especially on joint publications by partners. This underlines the improvement of dissemination of knowledge and a strong positive indication about the quality of the scientific results obtained in the project,
- Regular and strong increase of number of secondments of young researchers or students. This underlines the improvement of network attractiveness and of involvement on networking activities,
- Regular increase of the number of ASTEC users and of the number of stored experimental data. This underlines the improvement of integration through use of the reference tools (ASTEC, DATANET),

- Regular progress of access to the public web site. This underlines the improvement of network attractiveness.

1.4.2 Public website

Advanced Communication Tool (ACT)

In view of the large number of partners in the SARNET project, a tool for supporting document and information management as well as collaboration between partners was mandatory. In continuity of SARNET FP6, the ACT based on the Microsoft Sharepoint Portal Server was improved on diverse aspects: navigation, significant advances in search, detailed rights management, better user interfaces for handling (also large) files. The administration of users and user rights has been completely redesigned, based on a centralized Active Directory database containing the access information for all users. User groups were created for simplifying the rights management on work package level and for document libraries.

Setup, administration, maintenance and operation of the hardware and the system software have been performed within the framework of a subcontract to SfR (Solutions for Research). The subcontract includes also the public website, and has been ongoing for the full lifetime of the project.

In summary, the operation of the ACT has continued in a stable fashion, with high availability throughout the project's duration and satisfaction of partners. As the upload of project content (information and documents) is done by the participants themselves (in good part by the WP and sub-WP leaders), only little support effort has been needed to keep operation running smoothly. Some administrative activity was needed for managing users by means of the central Active Directory, and for managing users. Overall, the functionality of the ACT has provided a stable basis for document and information management as well as collaboration between partners for the whole lifetime of the project.

Public SARNET website (<http://www.sar-net.eu>)

The aim of the public website is to provide information about the project, its progress and results to an interested public by means of a well-structured, up-to-date and attractive site. Efforts were done to account for the feedback of SARNET FP6.

As a first step, the underlying software, which had proved too rigid to be easily handled by non-experts on website design, was replaced by an open-source product which has found wide diffusion, DRUPAL. Also, the former extension of the site, .org, has been replaced by .eu to indicate the affiliation with the European Union. The site's structure has been kept intact, the main themes covering Research Activities, Network Organization, Reference Documents, Events, Favourite Links, Partners and Contacts. However, the contents were thoroughly reviewed and significantly enhanced, particularly concerning the Homepage and Network Organization page containing essentials on the project; the description of Research Activities; the Reference Documents which now include publications in scientific journals, a list of presentations in conferences, and all papers and presentations of the ERMSAR conferences; the Events, Favorite Links, Partners and Contacts. By these measures, the website has strongly gained in content, actuality, attractiveness and visibility.

The effort of maintaining the actuality of the website was concentrated on the visibility of the project's progress, on new results generated within the project, and on information about events up to date. It was performed mainly by periodic newsletters, which reflect the current activities and recent findings.

1.4.3 Storage of experimental data

The operation of the DATANET experimental database which has been set up during the SARNET FP6 project, as based on the STRESA JRC tool, was continuing during the SARNET2 project.

The management and maintenance is performed by JRC/IET in Petten (The Netherlands). The JRC/IET team is ready to create new independent nodes in servers of the institutions interested in hosting them, as well as creating nodes based on an EC/JRC/IET address. This latter feature is a most convenient possibility to upload own data without the need for a trained specialist required for the set-up of an external-to-JRC node. This solution is also most convenient for members who would like to upload only small quantities of data. At the end of the project, data from 265 experiments in 43 facilities have been stored on 7 different partners' nodes.

JRC also offers training opportunities for the creation of new nodes or uploading data on JRC based nodes. In the same way JRC has provided Information Technology (IT) support within foreign laboratories directly, when needed (e.g. with AEKI in Hungary, KTH in Sweden).

An important development in the DATANET activities could be the fact that within a rather short term future the JRC IET is willing to undertake a full refurbishment of the data base using modern and recent IT tools, to guarantee a perennial use of STRESA for future years. This activity will be totally supported by the JRC IET, put into place and managed by its IT personnel. The timing of this support work is not linked to the SARNET2 project.

During the SARNET2 project, the following data sets have been stored in DATANET:

- By AEKI (now MTA-EK): CODEX experiments (degradation of VVER bundles), 8th series of RUSSET experimental programme (deposition of Ru oxides on different materials and transport of fission products). A mission by JRC personnel was organised in Budapest in early 2013 to repair the local node.
- By CIEMAT: data from the PECA facility (experiments done in the SGTR project).
- By IRSN: uploaded data from the following facilities, CAIMAN, EPICUR (iodine tests in 2010-2011, ruthenium tests in 2009-2010), PARIS (tests at AREVA GmbH), Phébus.FP (Minutes of the Steering Committee meetings and SAWG Status reports), SISYPHE, VERCORS, HEVA, MOZART, BECARRE, RUTHENIUM, EMAIC, CARAIDAS, CHIP Phenomenological and Analytical lines, TUBA, RTF, RECI (iodide particle conversion into gaseous iodine in contact with hydrogen recombiner plates).
 - There are experimental data files and also reports (objective of the experiments and experimental results). Several reports on the ISTP and OECD/STEM experimental programmes have been uploaded too;
 - Reports (general documents, objectives and experimental results) of the VERDON ISTP_1 test Quick Look report (ISTP DOC 145 and VERDON ISTP_2, 3 and 4 Test Objectives ISTP reports 147, 154).
- By KIT: DISCO-H series of experiments, LIVE-L6 data, QUENCH-16 and 17 data and reports.
- By VTT (data from FORTUM): data from the HORIZON I2-A tests.
- By JRC: apart from a large amount of JRC/Ispra based experimental data, the SARNET WP8-FASTG data on the Fukushima accident have been uploaded and stored between April 14 and June 10, 2011.
- By KTH: further to FOREVER, a report on DEFOR-A was uploaded. This independent node has become fully accessible with the technical support of JRC Petten IT team.

The CEA node needs more work before it can be operational. A new webmaster and responsible person has been nominated recently at CEA. Potentially data will be uploaded in the future on the following experiments: COLIMA, KROTOS, VITI, VULCANO, ARTEMIS, BALI, CORINE, SULTAN and TREPAM.

1.4.4 Spreading of excellence and knowledge

The Spreading of Excellence work-package was mainly planned to disseminate the knowledge in the SA field to young researchers and students by an education and training programme and by mobility grants. Furthermore it allowed to present the gained knowledge on SA through periodic “European Review Meeting on Severe Accident Research” (ERMSAR) which are becoming the major worldwide conference on SA research (the proceedings of these ERMSAR Conferences are available for download on the SARNET2 public web site). A second task for the dissemination of knowledge and information was through the publication of periodical newsletters and the participation to public events.

Inside the education and training program, the 1st SARNET2 Education and Training Course with the title “Severe Accident Phenomenology Short Course” was jointly organized during 1 week in January 2011 by University of Pisa and CEA, and hosted in Pisa by UNIPI. The participation reached about 100 students or young researchers from 20 countries worldwide, the most important participation among the successive courses in the SARNET FP6 and SARNET2 projects. It helped disseminating knowledge gained on SA in the last two decades to Master/PhD students and young/new researchers. The program covered SA phenomenology and progression in current water-cooled Gen. II NPPs, but also the different design solutions in Gen. III ones. The purpose was to describe Gen. III designs addressing SA (i.e. the “in-vessel” melt retention concept or the “ex-vessel” core catcher concept). The accident phenomenology has been described through its progression in the core and in the lower head up to vessel failure, followed by the ex-vessel accident progression, with the loadings which can cause early containment failure (i.e. Direct Containment Heating, hydrogen combustion in containment, steam explosion) and the late containment failure (i.e. MCCI, coolability, etc.). The source term with fission products release from the core and transport in the coolant system and in the containment has been specially emphasized. Lecturers were 18 experts from 8 different EU countries, with large skills and knowledge on Gen. III NPPs and on the progression of a SA. The presence of lecturers from industry was utilized to describe how the different plants would react during a SA, keeping in mind that it was an introductory course not allowing lengthy discussions or computer simulations. The course was open to Pisa University master students (11 students obtained the final participation diploma) and contributed for 3 ECTS as an advanced course for master/PhD students, with a strong link among SARNET2, ENEN & European Master of Science in Nuclear Engineering (EMSNE). The CD-Rom containing all the materials of the Pisa Course (PDF of lectures, movies, photos, educational notes from the first SARNET phenomenological course, materials from the draft of the SARNET SA text book) has been distributed to all the course participants and lecturers. Copies of this CD-Rom are available for dissemination, to be used also as education material.

The 2nd SARNET2 Education and Training Course with the title “Severe Accident Phenomenology and Management” was organized during 2 days in July 2012 in Karlsruhe by KIT with a strong involvement of AREVA GmbH and CEA, with a scientific committee helping to define the content and the speakers including IRSN and UNIPI. Following the recommendations of the SARNET2 Advisory Committee, it was a short 2 days course on general aspects of SAs and some general insights into SAM principles, without nuclear plant or country dependent features. The goal of this course was quite different with respect to the previous Pisa 2011 one because this 2nd course was mainly focused on disseminating the knowledge gained on SAs in the last two decades primarily to managers and senior scientists. The 2012 KIT course programme covered SA phenomenology and progression in current water-cooled Gen. II and III NPPs with the following main topics:

- historical overview including a review of TMI-2, Chernobyl and– above all - Fukushima accidents;
- short introduction to the SA phenomenology;

- overview of main SA codes (in priority ASTEC);
- SAM Guidelines;
- back-fitting of Gen. II NPP and SA mitigation for Gen III plants;
- radiological consequences to the environment and to the public resulting from a SA.

Lectures have been given by 10 international experts from major European nuclear institutes, industries and universities working on the SA topic. As expected, the participation at this course was good but lower with respect to the Pisa course, considering the particular high level target of the audience, with 32 participants from 11 countries worldwide.

The 3rd SARNET2 Education and Training Course with the title “Severe Accident Phenomenology Short Course” was held in the middle of April 2013 at ICL - Imperial College London (UK), organized by ICL, CEA, IRSN and UNIPI, with the participation of 58 young researchers or students from many countries over the world. This was again a 1-week course on SA phenomenology, following the strong requests to repeat this kind of course from different worldwide organizations after the Fukushima accident. The 2013 ICL course program, based on the skeleton of the Pisa 2011 Course, covered SA phenomenology and progression in current water-cooled NPPs, focusing on Gen. III mitigation solutions. It addressed also the recent status of knowledge about the Fukushima Daiichi accidents and an overview of the European stress tests.

At the beginning of 2012, the SARNET text book on SA phenomenology has been published by Elsevier with the title “Nuclear Safety in Light Water Reactors - Severe Accident Phenomenology”. It is a 700 pages book that covers in a complete manner the historical aspects of water-cooled reactor safety principles and the phenomena concerning in-vessel accident progression, early and late containment failure, fission product release and transport in RCS and containment. It contains also a description of reference analysis tools or computer codes, of management and termination of a severe sequence, as well as of the environmental management. This unique reference book emphasizes the prevention and management of a SA, in order to teach nuclear professionals how to mitigate potential risks to the public and the external environment to the maximum possible extent.

Three ERMSAR (European Review Meeting on Severe Accident Research) Conferences have been organized during the previous SARNET FP6 project successively in France, Germany and Bulgaria as an exchange forum for the whole international SA community. Following this “spreading of knowledge” approach, in SARNET2 the 4th ERMSAR Conference was hosted by ENEA in Bologna (Italy) on May 11-12, 2010. It gathered 98 participants from 23 different nationalities (8 non-EU countries and 15 EU ones). The network partners presented papers on their joint work, including some first conclusions of the work performed at the beginning of SARNET2. The final session was the opportunity for very interesting and lively discussions, in particular on the links with PSA2 and on the future of SARNET beyond FP7.

The 5th ERMSAR conference was the first one to be fully open to the international community. It was hosted by GRS in Cologne (Germany) on 21–23 March 2012, gathering 157 participants with 52 presentations, not only by SARNET partners. The participants came from 60 organisations and 27 countries (42 of them were from non-SARNET partners). This meeting aimed at presenting the current status of SA R&D both in Europe and elsewhere, as an “open conference”. Therefore, besides the presentations of current network by SARNET members, non-SARNET organizations were invited to contribute through papers and presentations to the conference. The SARNET members were also able to present SA individual work conducted outside of the JPAs. The four previous ERMSAR conferences provided insights on the SARNET network activities and in particular on the research priorities established in 2008. On the contrary, the global objective of this 5th seminar was to present the recent progress of international knowledge on SAs, of course including the work done within the SARNET network in the last two years. It was also an opportunity to

discuss future R&D priorities and, in particular, how the feedback of the Fukushima accident can be taken into account in planning the new research needs.

The preparation of the 6th ERMSAR conference, presenting the results of the whole SARNET2 project, hosted in Avignon (France) by IRSN from 2 to 4 October 2013, has intensively continued. Information on this Conference is available on the SARNET2 web sites.

The Mobility Programme (MOB) aimed at training young researchers and students through a delegation towards SARNET research teams, in order to enhance the exchanges and the dissemination of knowledge in the SA area. In this programme, the long term goal was to build and strengthen teams which would engage together in a certain activity of the excellence network. Thirty-three mobility actions, with an average duration of about 3 months, were completed at the end of the SARNET FP6 project, where most of the delegates came from Eastern Europe countries to laboratories and universities in Western Europe countries, a large fraction of the delegates were females and the dominant area of training was on the use of the ASTEC code. In SARNET2 an increased financial support covering partially the delegation costs (maximum of 2000 €/month) was provided to push these MOB actions. The number of these mobility actions is indeed one of the several integration indicators defined in order to assess the progress of the SARNET2 project and the success of the network integration and, in total 22 mobility actions were authorized. For the success of this MOB programme, strong partners' efforts were needed to highlight research projects which could be of interest to young researchers or students (the collected information was advertised on the private and public SARNET2 web sites) and to disseminate internally and externally the understanding of the benefits of this MOB programme. In this context an attractive idea has been to arrange, for the university delegates, to perform a master thesis and/or to take courses in SA technology in the European Master of Science in Nuclear Engineering (EMSNE) framework, funding the master thesis stages in the SA field to be performed to obtain the 20 credits necessary for the EMSNE achievement under the ENEN (European Nuclear Education Network) umbrella. After a slow start of the process with very few candidates, it accelerated from mid-time of the project with secondments in each WP. Among all mobility actions, there was no dominant topic except for WP8 with 9 secondments, a good balance between the two genders (13 M vs. 9 F) and average time duration of about 4 months. The origin of delegates was quite diverse, without the past predominance of Eastern countries. Differently from SARNET FP6, only 2 secondments on ASTEC training took place, which can be explained by the large progress of partners' experience on ASTEC use and the multiplication of ASTEC courses and users clubs.

The dissemination of knowledge and information was also performed through periodical newsletters available on the SARNET2 public web site: N°1 in October 2009, N°2 in April 2010, N°3 in October 2010, N°4 in June 2011, N°5 in January 2012, N°6 in October 2012, while the final SARNET2 newsletter N°7 will be released in January 2014.

Finally, a total of 101 papers in scientific journals (most of them peer-reviewed) and 262 publications in national or international conferences were released or presented in order to disseminate the information on the SARNET network activities and on the spreading of excellence programme. They are listed in details in the Section 2 (this does not include the 20 papers planned in the ERMSAR-2013 special issue of the Annals of Nuclear Energy journal that should be released around mid-2014). Among them, one can underline the following general publications:

- 1 general paper in the "Science and Technology of Nuclear Installations" journal in 2012,
- 8 general presentations in conferences: ICAPP, EUROSAFE, New Horizons workshop in India, NUSSA in China (the two last ones being invitations to present the SARNET activities out of Europe),
- 3 presentations on spreading of excellence activities (SNETP General Assembly, NESTet, CONTE 2013).

1.5 Conclusions on the SARNET2 project

After 4 years under the auspices of the 7th Framework Programme of Research and Development of the European Commission (and in continuation of the SARNET/FP6 project), the SARNET2 project has confirmed the large success of the SARNET network of excellence on research on severe accidents phenomenology and management:

- Development and optimization of competencies and resources in Europe,
- Efficient networking and integration of R&D activities of diverse types (experiments, modelling, computer codes) and from diverse types of organizations,
- Significant progress of knowledge through new experiments (in particular debris bed reflooding and MCCI), benchmarks between computer codes, and improvements of physical modelling,
- Development and reinforcement of the functionalities of the ASTEC integral code that has been confirmed as European reference,
- Storage of experimental data and reports in the DATANET database, using the JRC STRESA tool,
- Update of ranking of R&D priorities accounting for recent international R&D and, since 2011, for the impact of Fukushima accidents,
- Dissemination of knowledge through papers in journals and conferences, textbook, state-of-the-art reports, education and training courses, ERMSAR conferences....
- Collaboration with entities like OECD/NEA and US.EPRI.

It is difficult to mention all the gains in scientific knowledge obtained within the SARNET2 frame but the following main outcomes can be underlined:

- On corium and debris coolability:
 - Demonstration of possibility of effective cooling of debris beds by penetration of water, even for small debris. 2D and 3D effects were highlighted in DEBRIS and PRELUDE facilities, supporting the quantification of basic laws to predict the coolability behaviour under a wide range of conditions;
 - Support, for the first time, of external cooling of a VVER-440/V213 vessel lower head by large-scale experiments in the RESCUE facility, including one test simulating transition from a homogenous pool to a stratified MASCA-type pool;
 - Identification of two types of BWR vessel lower head creep failure, depending on control rod guide tube cooling modes and leading to different melt releases in terms of breach size, melt mass and melt compositions and superheat;
 - Joint OECD-SARNET benchmark on an alternative TMI-2 accident scenario that, in contrary to previous exercises, showed that the simulation codes are now able to calculate the accident sequence up to the more severe degradation conditions, including the core reflooding. The first important deviations in the results are observed after core geometry changes due to in-core melt progression and material relocation phenomena;
 - Analysis of spent fuel pool accidents for various types of reactors that contributed to complete the OECD state-of-the-art report and identified research activities necessary to reduce the uncertainties.
- On Molten-Corium-Concrete-Interaction:

- Some main lessons in dry conditions were drawn:
 - o First on the origin of discrepancy between the isotropic ablation of limestone-rich concretes and the larger lateral ablation than vertical ablation of silica-rich concretes, mainly due to the structure of pool/concrete interfaces;
 - o Secondly, for oxide/metal pools, on the basis of VULCANO experiments, phase repartitions which are different from simple-layers assumptions considered in MCCI codes and, in MOCKA experiments, a stable stratification but with a density contrast between phases typical only of the long term MCCI, as well a strong effect of iron bars in concrete leading to a more isotropic ablation;
 - o Thirdly, reactor applications showing no pool stratification with metal below in case of Limestone-Common Sand concrete and a late basemat melt-through exceeding 8 days but, in case of siliceous concrete, possible stratification and basemat melt-through only after a few days (despite many uncertainties remaining in the latter case).
- Mitigation measures by bottom and top flooding have been also considered and improvements of the COMET concept modelling have been proposed.
- On containment issues:
 - In the complex topic of fuel-coolant interaction, which may lead to steam explosion, reasonable understanding and modelling have been reached for a two-dimensional configuration with UO_2/ZrO_2 melts and a gravity-driven melt injection at the centre of the vessel;
 - The generic containment benchmark, in connection with hydrogen risk issue, clearly showed the uncertainties of results of accident simulations performed with lumped-parameter codes;
 - Computational Fluid Dynamics (CFD) codes were confirmed (despite needs of further modelling development) as an adequate tool for simulating specific phenomena such as influence of containment sprays, hydrogen combustion, steam condensation, or interaction between Passive Autocatalytic Recombiners and containment atmosphere.
- On source term:
 - Completion of two large code benchmark exercises on ThAI Iod-11 and Iod-12 tests. They showed that some thermal-hydraulics variables like gas temperature and pressure are well captured by codes, while a broad scatter was observed on relative humidity and condensation rates. Two major observations were made: the user effect was substantial and thermal-hydraulics largely affects iodine under the tested conditions. Additionally, a number of potential areas for improvement were identified, i.e. molecular iodine-steel interactions, iodine wash-down modelling and nodalization effect.
 - Demonstration that gaseous iodine species such as molecular iodine and volatile organic iodide react with the products of air radiolysis to form iodine oxide particles.
 - Proposal of a whole modelling of the evolution of airborne iodine in the containment that account for the interaction with paints, with paint degradation products and with air radiolysis products and for the stability of iodine aerosols including iodine oxides ones. The gas-to-particle conversion process could maintain the airborne radionuclide contamination of the containment at a higher level than would be obtained from simple equilibration of volatile species between the atmosphere and water volumes or surfaces. Any prediction of the long-term airborne iodine activity levels therefore needs to account for all the processes contributing to the steady state.

- Completion of a benchmark exercise on the Phébus FPT3 integral test that allowed to assess progress of simulation codes since ISP46 on FPT1 and to confirm the importance of the main on-going research programs.
- On ASTEC integral code:
 - Integration, in the V2 series of IRSN-GRS versions, of most knowledge generated by SARNET R&D through improved physical models proposed by the Topical WPs;
 - Code assessment by 30 partners (about 60 users), i.e. validation vs. experiments and benchmarks on plant applications (PWR, VVER...). Satisfactory (or even good) results were obtained for all phenomena except on reflooding of degraded cores, in particular the corresponding hydrogen production, and on pool-scrubbing. Benchmarks showed also the code applicability to most Gen.II-III NPPs, incl. EPR;
 - First calculations of a Limited Core Damage Accident in a PHWR, with validation on dedicated experiments (work done by BARC);
 - Large progress under way on core degradation models for BWR and PHWR that will be available in the next major version V2.1 planned end of 2014, as well as a new model for reflooding of a severely damaged core.

SARNET has become a “brand” that attracts in particular other non-European partners and new nuclear countries. But the momentum must not be lost, especially after the Fukushima accidents and the need to further consolidate nuclear power plants with new SA mitigation systems. After the end in March 2013 of the SARNET2 FP7 project, self-sustainability of the network is achieved through integration in the NUGENIA European association (www.nugenia.org) addressing Gen.II-III R&D, as one of the 8 technical areas. The SARNET2 update of research priorities has been totally used to build the NUGENIA Strategic Research Agenda on severe accidents. Networking activities will continue in this new framework in the same way as up to now, still in order to capitalize international knowledge and to contribute to a better prevention and mitigation of severe accidents in existing and future European NPPs of diverse types, thus to the improvement of their safety.

TABLE OF ACRONYMS

BWR	Boiling Water Reactor
CFD	Computational Fluid Dynamics
CRGT	Control Rod Guide Tube
CSNI	Committee on the Safety of Nuclear Installations
DBA	Design basis accidents
ERVC	External Reactor Vessel Cooling
FA	Fuel assemblies
FCI	Fuel Coolant Interaction
ISTC	International Scientific and Technical Centre
ISTP	International Source Term Programme
IVR	In-Vessel Retention
JPA	Joint Programme of Activities
LCS	Limestone Common Sand
LOCA	Loss Of Coolant Accidents (Small, Medium or Large)
LOOP	Loss Of Off-site Power
MCCI	Molten-Corium-Concrete-Interaction
MOX	Mixed OXide fuel
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
OECD	Organization for Economic Co-operation and Development
PAR	Passive Autocatalytic Recombiner
PWR	Pressurized Water Reactor
PHWR	Pressurized Heavy Water Reactor
RBMK	Reactor Bolshoi Moschmosti Kanaynyl
RCS	Reactor Cooling System
RPV	Reactor Pressure Vessel
SAM	Severe Accident Management
SARP	Severe Accident Research Priorities
SBO	Station Black-Out
SGTR	Steam Generator Tube Rupture
SFP	Spent Fuel Pools
SOAR	state of the art report
TLFW	Total loss of steam generator feed water
VVER	Water-Water Energetic Reactor