cesam

CODE FOR EUROPEAN SEVERE ACCIDENT MANAGEMENT

SP5-Euratom

Collaborative project

FP7 Fission 2012

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Final publishable summary report

1. Executive summary

The CESAM FP7 project of EURATOM has been conducted from April 2013 until March 2017 in the aftermath of the Fukushima Dai-ichi accidents. Nineteen international partners from Europe and India, including the European Joint Research Centre, have been participating under the coordination of GRS and with a strong involvement of IRSN. The project has been started against the background of the Fukushima accidents. These highlighted that both the in-depth understanding of such accident sequences and the development or improvement of adequate Severe Accident Management (SAM) measures are essential in order to further increase the safety of the nuclear power plants operated worldwide.

The project objectives were in priority an improved understanding of all relevant phenomena during the Fukushima Dai-ichi accidents and their importance for Severe Accident Management (SAM) measures and implementation in computer codes to simulate plant behaviour throughout accident sequences including SAM measures simulation. One starting step was the analysis of current SAM measures implemented in European nuclear power plants.

Model development has mainly been done for the ASTEC (**A**ccident **S**ource **T**erm Evaluation **C**ode) severe accident code that is continuously developed for the simulation of most relevant phenomena during severe accidents in the reactor of Nuclear Power Plants (NPP) and surrounding buildings. ASTEC has been jointly developed by IRSN (France) and GRS (Germany) from the late 1990s and is exclusively developed by IRSN today. Along with enhanced modelling of severe accident related phenomena, improvements have been implemented in the current ASTEC V2.1 series for the estimation of source term consequences in the environment and the prediction of plant status in emergency centres.

In order to achieve the project goals, simulations of relevant experiments that allow a solid validation of the ASTEC code against single and separate effect tests have been conducted. Covered validation topics in the CESAM project have been grouped in 9 different areas among which are re-flooding of degraded cores, pool scrubbing, hydrogen combustion, or spent fuel pool behaviour.

ASTEC reference input decks have been created for all reactor types operated in Europe today as well as for spent fuel pools. These reference input decks generically describe plant types like PWR, VVER, PHWR, and BWR without defining proprietary data of a special plant and they account for the best recommendations from code developers and users. In addition, a generic input deck for a spent fuel pool was elaborated. These input decks can be used as basis by all (and especially new) ASTEC users in order to understand code basic requirements and model features and to implement the specificities of their own NPP type. Based on these generic inputs, benchmark calculations have been performed with other codes (such as MELCOR, MAAP, ATHLET-CD, COCOSYS…) with a focus on applicability of ASTEC models to currently implemented SAM measures.

The technical work has been complemented by a special work package dealing with dissemination of project results to the interested public and performance of a mobility programme for young experts that allowed cooperation among different countries institutions.

2. Project summary description

Within CESAM, the main tool for analysis for accident scenarios and consequently the target for improved modelling development is the ASTEC computer code that is used for the simulation of most relevant phenomena occurring during severe accidents (SA) in the reactor of Nuclear Power Plants (NPP) and surrounding buildings. The ASTEC code has a strong modular approach allowing simulation of single-effect tests as well as of separate-effect tests in coupled mode with different modules activated up to integral plant applications enabling all modules that allow a comprehensive modelling of all relevant process from an initiating event up to the release of radioactive material into the environment. The modular structure of ASTEC is illustrated in Figure 1.

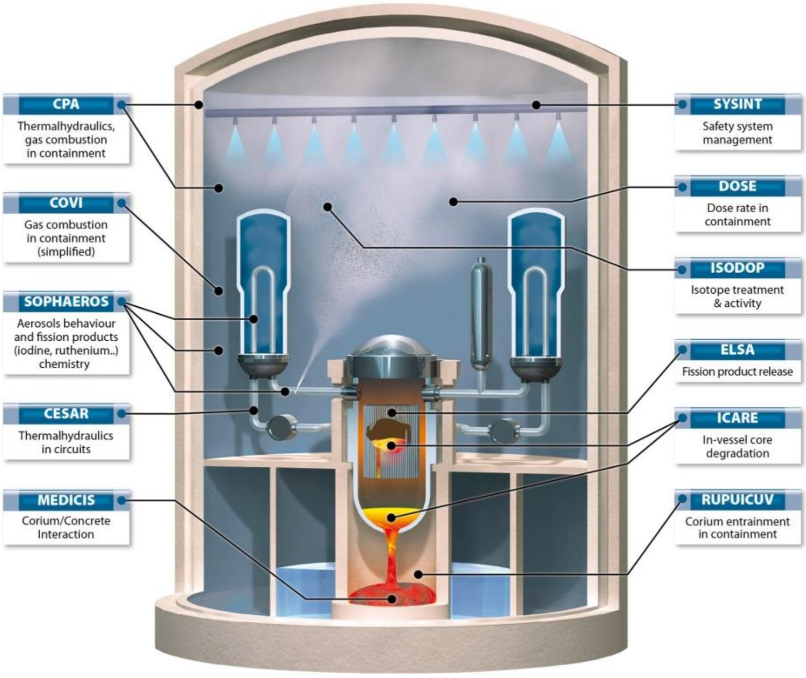

Figure 1 ASTEC V2.1 integral code - overview of main modules.

Different modules exist that simulate separate parts of the NPP respectively different phenomena and processes. Overall, four main domains are generally considered to represent a NPP. The reactor core and pressure vessel are modelled with the ICARE module, except for the thermal-hydraulics that is now also covered by the CESAR module which deals more generally with the whole reactor cooling system (RCS), the CPA module deals with the containment thermal-hydraulics simulation and the MEDICIS module is addressing Molten Core Concrete Interaction (MCCI) in the cavity after vessel failure. Further, separate modules are available for the simulation of different SA phenomena like SOPHAEROS, which simulates the transport and chemistry of fission products (FP) and aerosols (including notably iodine and ruthenium chemistry) in both the reactor coolant system (RCS) and the containment or RUPUICUV that evaluates the corium entrainment under elevated pressure causing Direct Containment Heating (DCH). Finally, accident management relevant systems like Passive Autocatalytic Recombiners (PARs) to remove hydrogen can be described (within CPA).

During CESAM life time two distinct series of ASTEC V2 major versions have been sequentially used: the V2.0 series during the first period of the CESAM project; and after the initial release of a “V2.1 CESAM only version” in March 2015 this series during the second part of the project. The CESAM project has been divided in three technical work packages on

1.) Assessment and validation of existing ASTEC models.
2.) Model improvements and implementation in subsequent ASTEC versions.
3.) Creation of plant input decks allowing calculation of severe accident scenarios with respect to SAM measures.

The first task on assessment and validation has been conducted in WP20 “Modelling assessment, improvement and validation (ASSES)”. Recommendations for code improvements were developed for all parts of the reactor and most modules of ASTEC first on basis of ASTEC V2.0 series. The identification and priority for such modelling improvements were driven by the phenomena considered as most relevant for SAM with respect to Fukushima Dai-ichi accidents first feedback. After being included in the ASTEC V2.1 new series by the ASTEC code developers in accordance to the CESAM recommendations, these code improvements have been again validated by the partners. During the first project period, reports on the availability of existing experiments (D20.21) and an analysis of missing models in the ASTEC code (D20.22) have been created. During the second project period, this work served as a basis for the validation of the at that time current ASTEC version V2.0rev3, documented in the validation report D20.24. Based on the outcomes of the V2.0rev3 validation and the analysis of model enhancement provided with the new ASTEC branch V2.1, specifications have been defined for models to be implemented in the upcoming ASTEC V2.1rev1 version in report D20.23. Within the last project period, the WP20 focus was on the validation of new ASTEC V2.1 branch, respectively the latest revision V2.1rev1 that was released about four months before CESAM end. The work cumulated in the final ASTEC validation report D20.25 “Synthesis of validation of ASTEC V2.1rev0 and rev1 versions” that can be seen as the ultimate outcome of assessment and validation work package.

The second task on model improvements and implementation in ASTEC has been performed within WP30 “Integration of models in ASTEC (INTEG)”. Different ASTEC patch versions including bug fixes have been released by IRSN all along the project lifetime in order to react to user requests on code applicability. The main work has been first on the achievement and then on the consolidation of the newly created ASTEC V2.1 series with the above mentioned release of the first version in March 2015. In order to promote the development of ASTEC to be used in support to decision making in SA scenarios, ASTEC V2.1 was also coupled to environmental consequences tools that simulate the distribution of released radioactive materials into the environment. At the beginning of the project the ASTEC version V2.0rev3 has been published that has been validated in WP20 and used for building of reference plant input decks in WP40. The new version ASTEC V2.1 has been distributed stepwise to CESAM partners. With a formal release of V2.1rev0 in March 2015 to CESAM partners the official deliverable D30.31 has been completed in April 2015. In the sub-work package on diagnosis the deliverables D30.32 “Interface package with atmospheric dispersion tools” and D30.33 “Methodology accounting for the uncertain information provided by the plant instrumentation” have been completed in 2016. The main efforts during the last project period have been first on the consolidation of the version V2.1rev0 for plant applications through the subsequent delivery of patch versions and then on the creation of the first revision of ASTEC V2.1rev1 in November 2016. The releases of these successive V2.1 versions have been of particular importance for the project, because they have been used in the WP20 validation and WP40 plant applications performed up to the end of CESAM.

The third task on plant input deck creation and accident calculations has been done in WP40 “Plant applications and Severe Accident Management (SAM)”. Generic reference input decks for the main types of NPPs (PWR, VVER, PHWR and BWR) operated in Europe have been prepared. During the CESAM project ASTEC has also been applied for analysis of accident scenarios in Spent Fuel Pools

(SFP). These reference input decks do not describe a specific plant, but can be used as example input datasets that show best practice for the development of datasets that simulate a specific plant. This generic approach is necessary to avoid the usage of proprietary plant data, but it allows the principal modelling of the features of the main plant types. Finally, these generic input decks have been used for plant analyses with a focus on possible improvements of ASTEC models for applications to SAM measures in various plant scenarios. At the beginning a report (D40.41) on the SAM approaches implemented in the different European countries and identification of related modelling requirements has been created. These recommendations have been directly used in combination with WP20 results for identifying, which models are relevant for simulation of SAM measures and should be improved or newly developed as part of WP30. Within the second project period, reference input decks for ASTEC version V2.0rev3 have been completed. The results for the first version of input decks have been summarized in deliverable D40.42 “1st set of “reference” NPP ASTEC input decks” in June 2015. Within the third project period work switched to the ASTEC version V2.1 and already existing V2.0 input decks have been converted to this new ASTEC series. Along with appropriate documentation these input decks have been completed in D40.45 “Set of final reference NPP ASTEC input decks”. Applications of these input decks with a focus on SAM measures have been summarized in D40.44 “Synthesis of evaluation of the impact of SAM actions through ASTEC NPP calculations” In addition a “Workshop on ASTEC capabilities for calculations of the Fukushima accidents, both in NPP and in SFP” (D40.43) has been coupled to the last CESAM technical workshop.

The technical work has been fostered by the work package WP50 “Dissemination of knowledge (DISS)” that coordinated the work among partners through yearly workshops and the setup of online websites for cooperation. In addition, the dissemination of project results through publications in conferences and journals has been performed. This way, the CESAM project supported the continuation of the active ASTEC user community. The public web site www.cesam-fp7.eu informs the scientific community and the interested public about activities and results of the project. In total 4 technical project workshops have been scheduled and done within the CESAM project:

- The first one hosted by IJS in Ljubljana, Slovenia in March 2014,
- The second one hosted by ENEA in Bologna, Italy in February 2015,
- The third one hosted by JRC in Alkmaar, The Netherlands in February/March 2016, and
- The final one hosted by KIT in Eggenstein-Leopoldshafen near Karlsruhe, Germany in March 13th to 15th 2017.

A BWR Fukushima workshop has been embedded within the last Workshop at KIT, where BWR relevant work of the CESAM community has been presented to a broader audience and discussed there.

In addition, the four editions of the ASTEC Newsletter were published on the project web page and also distributed by email as pdf through all CESAM partners and it was also sent to the ASTEC user community. Finally, through implementation of a CESAM mobility programme two PhDs from KIT and RUB went to IRSN-Cadarache for a 6-month delegation period during the second project period. Then, during the last project period, a BARC researcher stayed at IRSN-Cadarache too for a 5-month delegation period within the ASTEC development team.

3. Project main results

The following subsections describe in detail the work performed and the main results achieved on ASTEC model improvements, especially in the V2.1 series, validation, and plant applications with respect to SAM as well as the work of the ASTEC user's group supported by the dissemination of knowledge part of the project. The description is to a great extend taken from already created ERMSAR contribution by Nowack [NOW17].

3.1 ASTEC basic evaluation and model improvement/development

At the beginning of CESAM, the ASTEC version V2.0rev3 was released and evaluated. Based on the users' feedback, recommendations for code developments towards the V2.1 series have been made that have been partially considered in the CESAM work on model improvement and validation. During CESAM life time a switch has been performed from the ASTEC V2.0 series to the V2.1 series that was not completely driven by CESAM work but general requirements were addressed by the IRSN development team. In comparison to V2.0, the main new features of the V2.1 major version concern a strong re-engineering of the in-vessel coupling technique between the RCS thermal-hydraulics module CESAR and the core degradation module ICARE, and also significant evolutions of the SOPHAEROS module to enlarge its scope of application (FP/aerosols transport and chemistry) to the containment domain. The V2.1 version also included new core degradation models specifically addressing BWR and PHWR reactor types in order to make ASTEC able to better cope with some specific in-vessel processes supposed to occur in these two kinds of plants and thus to make ASTEC applicable in the end to complete SA sequences analyses for any generic type of NPP operated in Europe. Besides, several other physical modelling improvements have been implemented in the code by IRSN in two steps: firstly within the V2.1.0 version that was delivered in March 2015 to allow CESAM partners starting their tasks relating to either the code assessment (see Section 3.2) and/or the realization of full scale analyses and plant benchmarks (see Section 3.3); later on within the V2.1.1 version that was delivered in November 2016 to allow CESAM partners completing these tasks using an updated version. Furthermore, during the progressive elaboration and internal testing of the V2.1.1 new version, several ASTEC V2.1.0 patch versions have been successively released by IRSN from mid-2015 to mid-2016 to answer as fully as possible to most users' requests, the latest one V2.1.0.5 acting as a further consolidated version for plant complete applications (see Section 4).

To give some better idea of the V2.1.1 overall capabilities, few more comments about the main modelling improvements that are now available in this version with respect to V2.0rev3 are given hereafter. More details of the ASTEC V2.1 improved modelling can be found in Chatelard et al. 2016 [CHA16].

Looking first at the ASTEC V2.1 main structuring evolutions, one can set the following remarks:

- The CESAR/ICARE new coupling, that aimed at assigning more distinct roles for the two modules (thus allowing suppressing the rather sensitive CESAR-to-ICARE switch that was inherent to the operation of the V2.0 former series), constitutes the backbone of the in-vessel phenomena description in ASTEC V2.1 for complete plant applications. Moreover, a 2D extension of CESAR was simultaneously developed to support a radial discretization of the core region as in ICARE, thus allowing accounting for in-core 2D two-phase flow patterns.

- As to the ICARE modelling evolutions relating to BWR and PHWR cores, they have notably consisted in developing new dedicated components to properly represent the BWR canister walls and cruciform control blades as well as the in-core multi-channel flows. Furthermore, some other modifications have been made in the CESAR module, to deal e.g. with horizontal channels in the peculiar case of PHWR cores.
- Besides, a new component has been recently developed in CESAR to allow modelling of an external reactor vessel cooling (ERVC), along with an implicit thermal coupling between the vessel domain and such an ex-vessel cooling circuit.
- Concerning SOPHAEROS, the aim of extending its scope of application to the containment was to harmonize everywhere in ASTEC the treatment of FP/aerosols behaviour, thus naturally providing a more consistent overall modelling with respect to the source term evaluation.

As to the physical modelling relevance, efforts have been notably focused in the V2.1 series on the following topics:

- Reflooding of severely damaged cores: new dedicated model to specifically address the quenching of a porous media (debris bed geometry) that is representative of the late degradation phase;
- Oxidation of Zircaloy claddings under air atmosphere: speed-up of the oxidation kinetics when fuel rod cladding is exposed to a O_2/N_2 mixture at low temperature; formation of nitrides when nitrogen is in contact with the cladding at high temperature; and oxidation of nitrides, but only according to kinetics data to be necessarily supplied by users in ASTEC input decks;
- Corium behaviour in the lower head after its slumping from the core: upwards radiative exchanges between this relocated corium and the core support plate bottom face, up to its melting or collapse; new functionality in the original core magma oxidation model to allow it now dealing also with the oxidation of a corium pool once formed in the lower head; new features to better address in-vessel melt retention strategy (behaviour of corium layers and crust in the lower head, modelling of a typical ERVC…);
- Ex-vessel corium coolability during MCCI with a focus on the MEDICIS top quenching models addressing the corium coolability by water injection from above: better heat transfer evaluation at the corium top interface in presence of water, accounting for a complete boiling curve; account for a dedicated debris bed layer above the corium layers and apart from the upper crust; integration of new models for describing melt eruption and water ingression phenomena;
- Source Term evaluation capabilities in both RCS and containment:
    - Extension to elements Mo and B of the RCS model for gaseous phase chemistry kinetics, thus allowing SOPHAEROS to address the Cs-I-O-H-Mo-B system;
    - New module for modelling of pH behaviour in sumps; continuous improvement of the iodine chemistry modelling to keep it close to the state of the art (new iodine-paint model, new reaction between gaseous organics and gaseous I_2 with possible formation of gaseous CH_3I by a radiolytic process, radiolytic decomposition into gaseous I_2 of iodine oxides and multi-components aerosols coming from the circuit such as CsI, CdI_2, AgI…, account for the volatility and reactiveness of gaseous HOI, improvement of the model for interaction of gaseous iodine and steel surfaces…).

In addition to the general ASTEC development and release of subsequent versions and documentation, the applicability of ASTEC in support to decision-making has been enhanced. The work consisted on one hand on building-up a fully chained computation tool to evaluate environmental consequences of a SA. This has been achieved at IRSN by implementing a general interface of ASTEC V2.1 to produce source term data that could be used by environmental consequence tools as input data to calculate then the distribution of released radioactive materials in the SA course. While the data between ASTEC and the consequence tool are transferred through adequate files, a dedicated Graphical User Interface was set-up by IRSN to manage other information exchanges relating notably to the input creation, such as e.g. the selection of ASTEC result files, the meteorological data etc. On the other hand a method based on Bayesian Networks (BN) to assess the likelihood of the different possible accident scenarios given the uncertain information provided among others by the plant instrumentation has been implemented.

As to the radiological consequence part of the tool, the programming of the interface has been achieved at IRSN in 2016, and then separately tested by both IRSN and GRS. The main tasks assigned to this tool are:

- To transform the source term data (provided by ASTEC) into any atmospheric dispersion code tool complying with the IRIX standard format exchange;
- To fetch and eventually transform other necessary information for the consequence calculation (meteorological data…);
- To launch the ASTEC/dispersion tool or a dispersion tool alone, using an already existing ASTEC calculation if available;
- To read the results from the dispersion tools' calculation output and to return them to the Graphical User Interface.

It has been chosen to implement the coupling of the tool based on the IRIX exchange information standard. At this time, even if RODOS and pX dispersion tools, developed by KIT and IRSN respectively, are planned to be compatible with this format, none of them are yet operational with this file format.

As to the diagnosis part of the tool, the work has been performed in the framework of a French PhD thesis co-mentored during the CESAM lifetime by IRSN and Paris VI University. Main objectives of the PhD were the investigation of interests and limits of BNs in the field of nuclear SA, and more particularly to investigate whether the diagnosis capabilities of ASTEC in case of accident could be improved using both the development of uncertainty theories and the extended current knowledge of SA phenomenology. The future use of ASTEC for support to diagnosis is indeed to be based on BNs, keeping in mind that the most important contribution to diagnosis of such tools is that they can work backwards (diagnosis and inference) and forward (prognosis).

Concretely, the PhD thesis provided a basis for a tool that shall be able to intake onsite or offsite measurements or user-given information [MAB14]. In particular, since issues related to the use of deterministic codes have been seen to be of importance, a new approach was proposed to address the problem of deterministic dependences between random variables modelling the various phenomena occurring in a damaged NPP during a SA. The second main issue is related to the fact that accident simulation codes generally use continuous attributes, while inference is easier with discrete variables. To solve this second issue, a new clustering-based approach for multivariate discretization was proposed, that takes into account the conditional dependences among variables discovered during the BN learning. This proposed algorithm has been successfully tested at IRSN on random generated BN, exhibiting very promising results in comparison to existing algorithms [MAB15].

The next step, currently underway at IRSN, consists in the production with ASTEC of a calculations data base. The specifications of the calculations have been made so that a sufficient amount of calculations is available, but preserving a certain level of detail in the representation of the accident. Then, using the BN modelling the accident, the integrated tool shall present a selection of adequate ASTEC calculations matching the observations, so that the selected calculations of interest could be restarted from the accident present time.

After release of the initial V2.1 version, work switched to validation of this new series of ASTEC development, what is summarized next.

3.2 ASTEC assessment and validation

Within the assessment and validation it has been checked if all models available in ASTEC are at the state-of-the-art or if additional modelling efforts are necessary to simulate specific effects. In addition the validation of implemented models against existing experiments was performed. The selected topics represented all parts of the NPP and have been divided in nine categories:

- Re-flooding of degraded cores
- In-vessel corium retention via external reactor pressure vessel cooling
- Coolability of corium/debris in the cavity after reactor pressure vessel failure
- Behaviour of hydrogen risk mitigation devices
- Consideration of different filter systems for containment venting
- BWR in-vessel core behaviour
- Steam/aerosol release into water pools, pool scrubbing, resuspension
- Intentional depressurization of the RCS
- SFP behaviour.

The project starting with the assessment and validation of the at that time current ASTEC version V2.0rev3 that is reported in D20.24 "Synthesis of validation of ASTEC V2.0rev3 version" [BUC15]. After availability of the new V2.1 series of ASTEC work switched to the validation of this series with complete new modelling as explained in section 3.1. The work is included in the final validation report of CESAM D20.25 "Synthesis of validation of ASTEC V2.1rev0 and rev1 versions" [BUC17]. The work performed by the different partners in this document is summarized in Table 1.

Table 1 Overview of work performed within report D20.25 as given in [BUC17].

Task	Contributor
Re-flooding of degraded cores	
LOFT LP-FP-2 including reflood phase	ENEA
TMI-2 accident including reflood phase	ENEA
Bundle degradation and reflooding test QUENCH-08	KIT-INR
Bundle degradation and reflooding test QUENCH-11	RUB
PRELUDE 1-D and 2-D tests on quenching of hot debris	IRSN
DEBRIS tests on quenching of hot debris	USTUTT
In-vessel corium retention via external reactor pressure vessel cooling	

Task	Contributor
Thermal load estimation for VVER-1000	IVS
IVMR strategy with external water cooling for VVER-1000 reactor type	INRNE
RESCUE-2 experiment on two-phase heat flux from vessel external surface	IVS
BALI experiments on molten pool heat transfer	USTUTT
COPO II-Lo experiment on molten pool heat transfer	VTT
Coolability of corium/debris in the cavity after reactor pressure vessel failure	
Sensitivity calculations of MCCI test case with top cooling for VVER-1000 reactor type	INRNE
MCCI experiments in dry conditions with reinforced concrete – MOCKA 7.1 and 5.7 tests	GRS
MCCI experiments in dry conditions with prototypic materials – CCI-2, CCI-3 and VULCANO VBU5 tests	IRSN
MCCI experiments with top cooling – CCI-7 and CCI-8 tests	IRSN
Experiments on coolability of ex-vessel particulate debris – COOLOCE tests	VTT
Direct Containment Heating (DCH)	
Validation on DISCO experiments	IRSN
Behaviour of hydrogen risk mitigation devices	
Validation on experiments with NIS PAR	NUBIKI
Validation of CPA FRONT model	IJS
Estimation of source term	
Validation of the ASTEC modelling relating to the aerosols/FP transport in the circuits	IRSN
Validation of the ASTEC modelling relating to the iodine behaviour in the containment	IRSN
Validation of the ASTEC modelling relating to the thermalhydraulics and aerosols/FP transport coupling in the containment	IRSN
Validation of the CPA model against PASSAM experiments	CIEMAT
BWR in-vessel core behavior	
Experiences gained during creation of BWR reference input deck	GRS
In-pile bundle degradation test DF-4	USTUTT
Bundle degradation and reflooding test CORA-17	RUB
Steam/aerosol release into water pools, pool scrubbing, resuspension	
Review of CPA-SPARC modelling capabilities	CIEMAT
Intentional depressurization of the reactor cooling system (RCS)	
Validation against PACTEL T2.1 experiment	IVS
Spent fuel pool (SFP) behavior	
Summary of practices for modelling of SFP accidents with ASTEC V2.1 code gained in air-SFP NUGENIA+ project	IVS
Application of ASTEC V2.1 to single test of OECD-SFP project 1x1	CIEMAT
Comparison of ASTEC modelling capabilities with CFD tools	JRC

In the following, two examples are to be given. The first one is a specific application in the area of re-flooding: simulation of the QUENCH-08 test. The second one has a broader scope in the area of aerosol scrubbing in pools: validation of jet scrubbing against the available database.
The simulation of the QUENCH-08 re-flooding test has been conducted by KIT, Germany (the experiment was also done by KIT). The experimental conditions and results are described by Stuckert 2008 [STU08]. An overview about all QUENCH experiments is given in Steinbrück 2010 [STE10]. A horizontal cut through the used test bundle is given in the left part of Figure 2. The test bundle is composed of 21 fuel rod simulators (FRS), each one with an approximate length of 2.5 m. Twenty of these are electrically heated over a length of 1024 mm through a 6 mm diameter tungsten rod, whereas the unheated one is placed in the centre of the test section. Each heated FRS is composed of a tungsten centre rod, which is surrounded by ZrO2 annular pellets and a Zry-4 cladding. The tungsten rods are in contact with molybdenum and copper electrodes at the top and at the bottom, both electrodes being connected through cables to the DC electric power supply, and coated with a ZrO2 fibre. The unheated FRS is composed of ZrO_2 pellets and Zry-4 cladding along its entire length. All FRSs are fixed in their positions by five grid spacers (lowest with Inconel, the rest with Zircaloy). Four corner rods are installed to obtain a uniform temperature distribution of the rods. The corner rod B is removable and allows measuring the oxide layer axial profile before reflooding starts. The FRSs and the corner rods are surrounded by a shroud, which is at the same time surrounded by a tubular cooling jacket. The gap between the shroud and the cooling jacket is filled with a porous ZrO_2 fibre insulation (along the heated length of the bundle) and with stagnant argon gas above the heated length (1024-1300 mm). The cooling jacket is argon-cooled along the heated length of the bundle, and water-cooled above the heated length of the bundle.

In order to evaluate ASTEC V2.1 capabilities to simulate the QUENCH-08 test, the predicted cladding temperature histories at representative bundle elevations and the axial temperature and oxide profiles at representative times have been compared to the experimental data [**GOM17**]. In turn, both histories affect the hydrogen generation (see right part of Figure 2), which is correctly predicted both during the test (76 g ASTEC vs 84 g exp.) and during reflooding (30 g ASTEC vs 37 g exp.). The deviations seem to arise from the lower temperatures predicted at the upper part of the active zone (800-1024 mm) and the lack of cladding failure (and hence absence of molten material oxidation), both being a consequence of the radial temperature underestimation that also leads to a shroud temperature overestimation.

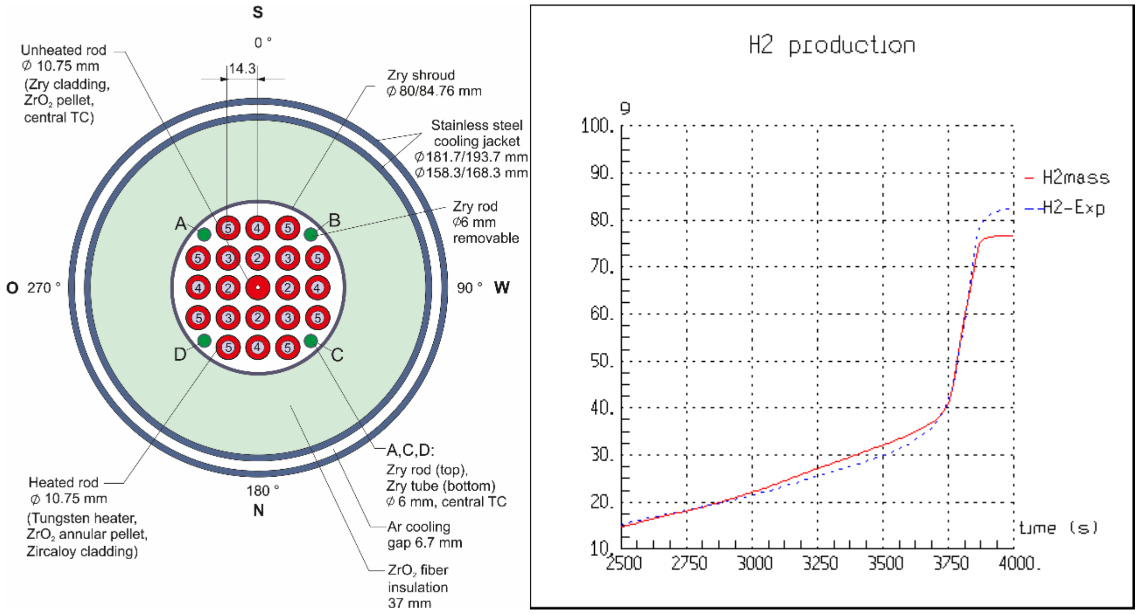

Figure 2: Left: Horizontal cut of QUENCH-08 bundle [STU08]. Right: H2 production in QUENCH-08 experiment [GOM17].

The general trends of this analysis show the capabilities of ASTEC V2.1 to reproduce all experimental phases of the QUENCH-08 test and hence, the key phenomena governing the SA early in-vessel phase. In addition, sensitivity analyses highlight the sensitivity of the hydrogen to the axial discretization, the outer electrical resistance of the bundle, as well as to several test boundary conditions.

As for pool scrubbing validation, thorough generic validations of ASTEC SPARC versions based on SPARC-B/98 [FIS98] had been performed previously by Herranz et al. 2009 [HER09] and at the beginning of the CESAM project [BUC15, section 6.2]. The main insight gained from those studies was that pool scrubbing models in ASTEC under-predicted absorption of particles in the aqueous bulk of pools, and such under-estimate was particularly large at high DFs (Decontamination Factors, ratio of the particle mass entering the pool to the mass leaving it).

In addition, CIEMAT intended to validate a specific part of pool scrubbing modelling in ASTEC: aerosol retention at the injection nearby when particles are carried by a gas moving at very high velocities (i.e., jet injection regime). To do so, a specific matrix for “jet scrubbing” validation has been set up based on the existing pool scrubbing database (Herranz et al. 2014 [HER14]) by choosing those experiments that met conditions of gas jet injection and low submergence (the tests belonged to different programs, like ACE, RCA, POSEIDON, etc). The sample consisted of a total of 18 tests, 12 of which contained no/low steam concentration in the carrier gas. A 3-zone ASTEC model was built in which the pool was included in the second zone, the first one being a buffer volume to modulate the injection conditions and the third one was the environment. By defining the DF ratio, RDF, as

$$R_{DF} = \frac{DF_{DATA}}{DF_{CPA}} = \frac{m_{CPA}^{out}}{m_{DATA}^{out}}$$

one may get information on how many times the mass leaving the inlet region estimated by CPA, either to the pool atmosphere (shallow pools) or to the bubble rise region (deep pools), is higher than what actually measured. Figure 3 shows that, for low and moderate DFs (i.e., DF < 20; scrubbing efficiencies lower than 95%), CPA overestimates the outgoing mass from the inlet region in less than a factor of 3.0 over data. However, when actual scrubbing efficiency is higher than 98% (DF ≥ 50), the mass that CPA estimates moving out the inlet region turns out to be more than 30 times higher than observed and this factor might be substantially higher if retention grows (except for one out of 12 tests represented). In other words, one may conclude that there seems to be a sort of systematic under-prediction of data on decontamination in the pool inlet region at high gas injection velocities and such under-estimate is even more noticeable when in-pool retention is massive. Nonetheless, the ASTEC deviations reported are a consequence of the state of the art on the matter, and not a specific ASTEC feature. A lot of work remains to be done in this field, though.

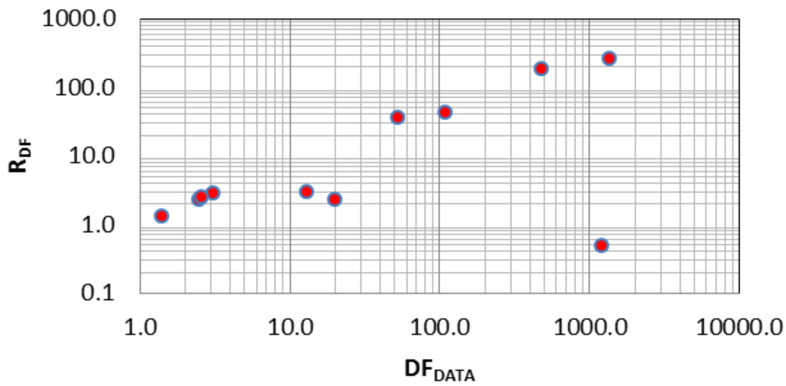

Figure 3: DF ratio vs experimental DFs for "non-steam tests"

All the calculations described in the final validation report D20.25 "Synthesis of validation of ASTEC V2.1rev0 and rev1 versions" [BUC17] were carried out with the ASTEC versions V2.1.0 and V2.1rev1 and mainly addressed issues related to accident mitigation measures, realistic estimation of the source term and extended applicability for BWR and SFP.

With respect to the re-flooding of degraded cores, the capabilities of ASTEC V2.1 was assessed against the reflooding phases in the LOFT LP-FP2 experiment and the TMI-2 reactor accident. For validation tests in (degraded) bundle configuration test (QUENCH-08 and -11) as well as in particulate debris configuration tests (PRELUDE and DEBRIS) have been calculated.

In-vessel retention via external cooling was assessed for the VVER-1000 reactor type and was further addressed in the validation calculations on BALI and COPO-II-LP experiments (corium pool thermal behaviour) and on the RESCUE-2 experiment (external cooling).

The coolability of corium/debris in the cavity after vessel failure was assessed in sensitivity calculations for a VVER reactor case with top cooling. Validation cases comprised MCCI in dry conditions (MOCKA 7.1 and 5.7 test, CCI-2 and 3 and VULCANO-VBU5) as well as under top flooding conditions (CCI-7 and CCI-8 tests). The coolability of particulate ex-vessel debris was addressed by validation calculations on COOLOCE experiments.

The issue of direct containment heating was investigated through validation calculations on DISCO experiments.

With respect to the behaviour of hydrogen risk mitigation devices the modelling in ASTEC has been assessed versus experiments with NIS passive autocatalytic recombiners in the THAI model containment. Further, the flame front progression model in ASTEC V2.1 was validated against THAI HD experiments.

The capabilities of ASTEC-V2.1 concerning the estimation of the source term were validated with respect to the aerosols/FP transport in the circuits (TUBA-D07, FALCON-18 and STORM SR11 experiments), the iodine behaviour in the containment (PHEBUS-RTF1 and CAIMAN 97-02 experiments), the coupling of thermalhydraulics and aerosols/FP transport in the containment (VANAM-M3) and also against experimental results from the PASSAM project.

Concerning the modelling of BWR in-vessel core behaviour assessments were carried out during creation of BWR reference input deck and the experiences were transformed into recommendations. Experimental validation cases comprised the in-pile bundle degradation test DF-4 and the bundle degradation test CORA-17, which also included a reflooding phase.

With respect to steam/aerosol release into water pools, pool scrubbing and resuspension the SPARC model has been reviewed on the basis of a large validation matrix containing many tests from RCA, EPRI, POSEIDON and ARTIST-II experiments.

Concerning the capabilities of ASTEC V2.1 to simulate the intentional depressurization of the reactor cooling system as accident mitigation measure validation calculations against PACTEL T2.1 experiment were carried out.

Finally, the applicability of ASTEC for the analysis of spent fuel pool behaviour was investigated.

In general, the results in the final validation report confirm the applicability of ASTEC V2.1 also to accident sequences and scenarios involving accident mitigation measures. Significant improvements have been reached with the latest ASTEC version V2.1rev1 especially concerning the coupled description of thermal-hydraulics and core degradation, now extending farther into the late phase and addressing also the quenching of core debris. As far as addressed by the limited studies in the project, the assessment and validation results for the widened simulation scope, now also including e.g. in-vessel BWR degradation and thermal-hydraulics as well as SFP behaviour, point out that ASTEC V2.1 is developing into a viable tool also for these fields. Respective further needs for development and improvement have been worked out in the final validation report D20.25.

3.3 ASTEC plant applications with respect to SAM

Within the ASTEC plant application work, reference input decks for different types of NPPs operated in Europe and India have been created and also modelling of a generic SFP has been worked on. In total seven working groups related to different reactor types have been built that typically consisted of multiple CESAM partners. These groups have created input decks for German Konvoi PWR, French PWR 900, VVER 440/213, VVER 1000, GE BWR4-Mark I, PHWR, and a generic SFP. An overview of the partners involved in the different reactor groups is given in Table 2. These generic reference datasets can be used by all ASTEC users in future as starting point for understanding main features and model requirements of ASTEC and for the adoption of their own to plant specifics. Therefore, they have been built up to include core, reactor vessel and cooling system, and containment in the nodalisation of the various plant types, but they avoid using proprietary real plant data. These datasets have also been used to perform accident scenarios with the special focus on the overall effectiveness of selected SAM measures.

Table 2 Overview of input deck creation

Nr	Reactor type	D40.42/ D40.45	Partners
1	French PWR900-LIKE	X/X	IRSN, AREVA, ENEA, JRC
2	German KONVOI 1300	X / X	GRS, USTUTT, RUB, KIT
3	VVER-440	X / X	IVS, NUBIKI, VUJE
4	VVER-1000	X / X	INRNE, BARC
5	BWR	- / X	GRS, CIEMAT, LEI, JRC, VTT
6	PHWR (CANDU)	- / X	BARC

In the beginning of the project an overview has been created what are the SAM approaches implemented today in different European countries and what are the modelling requirements to describe the situations of different SAM measures with SA codes. The results have been published by Hermsmeyer et al. 2015 [HER15].

During CESAM, generic input decks were first created on the basis of the previously existing ASTEC 2.0 version that were included in the deliverable D40.42 "1st set of "reference" NPP ASTEC input decks" [CHA15]; this phase allowed consolidating input decks and creating, in some cases, equivalent input decks in codes like MAAP and MELCOR for later code comparisons. The BWR and PHWR reactor models are two exemptions to this approach, because they both use geometry definitions that were made available only with ASTEC V2.1.

With the release of the new ASTEC V2.1 version the working groups could transfer V2.0 input decks and start running scenarios for verification and validation purposes. The corresponding input decks have been described in the CESAM deliverable D40.45 "Set of final "reference" NPP ASTEC input decks" [CHA17]. This extensive testing by a community of users, and for a wide range of models, and the related effort of rapidly improving the robustness of the new version is one of the key achievements of CESAM. Code assessment for the plant generic input decks has been a desktop exercise of checking simulation results for plausibility. Given the changes introduced with the new version, even substantial changes in the timing of events in accident scenarios as compared to V2.0 of ASTEC are possible. This holds also for comparisons with other codes, where the origin of differences can be in the code itself, but could also lie in different model abstractions in the input decks for different codes. Results of severe accident calculations using the latest ASTEC V2.1 series have been described in the report D40.44 "Synthesis of evaluation of the impact of SAM actions through ASTEC NPP calculations" [LOM17].

As an example, IRSN has documented, for the PWR 900 generic reactor model, and for ASTEC V2.0, a thorough analysis of a 2" Small-Break LOCA scenario in the cold leg, with unavailability of systems feeding water into the primary circuit, including safety injection - passive accumulators are assumed available. In this scenario, power supply is generally available. It is assumed that operators stop RCS main pumps at a certain point in time (1,200 s after SI failure) and start regulating Steam Generators (SG) to 33% of the narrow range water level to stop them from getting completely filled with water. For the system response it is also important, that this generic model contains a logic of automated responses that will start the Containment Spray System (CSS) once the containment is pressurised to 0.24 MPa. PARs installed in the containment to remove hydrogen are available.

The results comparing ASTEC versions 2.0 and 2.1 are somewhat similar in this special case. In contrast, Figure 4 shows a comparison of ASTEC V2.1 results with MAAP, in terms of primary and secondary pressures. The primary pressure is following the secondary pressure until, at 3,000 s, the effect of water depletion in the core and of loop clearance opening a gas path from the core to the break lead to a steep pressure drop. The primary circuit accumulators discharge from 4.28 MPa and are isolated when the pressure falls below 1.5 MPa, which causes that the isolation is much later in the MAAP calculation. ASTEC and MAAP behave indeed quite differently during the accumulators discharge phase for the selected SBLOCA sequence. As a consequence, differences between ASTEC and MAAP in primary pressure evolution and e.g. vessel rupture (ASTEC: 18,500 s, MAAP: 27,800 s) are significant – they do not rule out one of the results, but they require further analysis that goes beyond the point of the present paper. Only the general trends and the timing of the core relocation into the lower plenum (around 15,000 s) agree well.

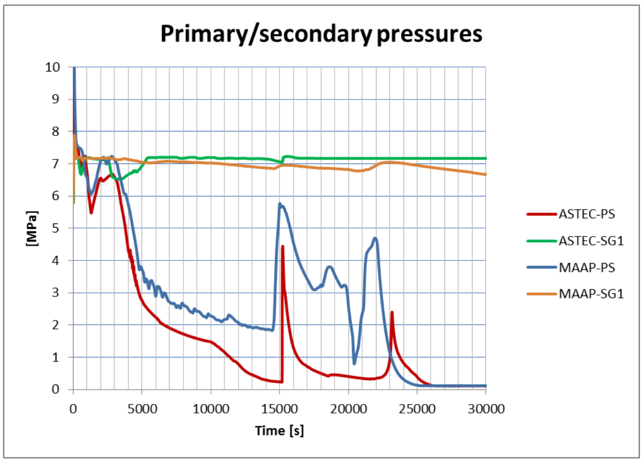

Figure 4: Comparison of SBLOCA simulation in ASTEC V2.1.0.5 with MAAP V5.0.2

3.3.1 Effectiveness of SAMG actions

The CESAM project had the goal of improving the modelling features in ASTEC with regard to SAM measures. This reflects the recommendation of the EU stress tests that SAM needs to be improved and that this goal can be supported by SA codes that help understanding the impact of SAM actions. It has to be stressed that codes are a complement to, but do not replace, the role of expert judgement, exercises, and training.

While a wide range of these SAM measures can be modelled with the ASTEC V2.1.0.5 version used at the end of CESAM, some phenomenological models are still in the process of completion. As an example, late core reflooding physical model has been improved in the V2.1.1 version, while still needing to be further consolidated in order to become more reliable. Such consolidation works are currently underway at IRSN in order to make this new promising model fully usable at plant scale.
An important goal of CESAM has been to simulate and document scenarios that demonstrate ASTEC's capability of investigating the effect of SAM actions, thus complementing the CESAM database of generic plant input decks with sample applications that support the ASTEC use. Almost all partners are applying the code with this purpose, and have fixed early in the project the scenarios

that they are investigating. These are mostly SBO and LOCA sequences, with various SAM actions available for the different reactor types. Where partners had access to other SA codes, the system responses have been compared. SFP and their relevant SA scenarios have been developed in a similar fashion.

The following example takes the above mentioned PWR 900 SBLOCA simulated by ASTEC as a base case (A) and compares it with two scenario variations, knowing that the core exit temperature set point initiates the application of SAM measures:

(B) at 4290 s, the operators initiate maximum cooling of the steam generators
(C) after 2 hours the Low Pressure Safety Injection (LPSI) is recovered; in fact, it only becomes active when primary pressure falls below 1 MPa. Figure 5 above displays the results of ASTEC for the three cases.

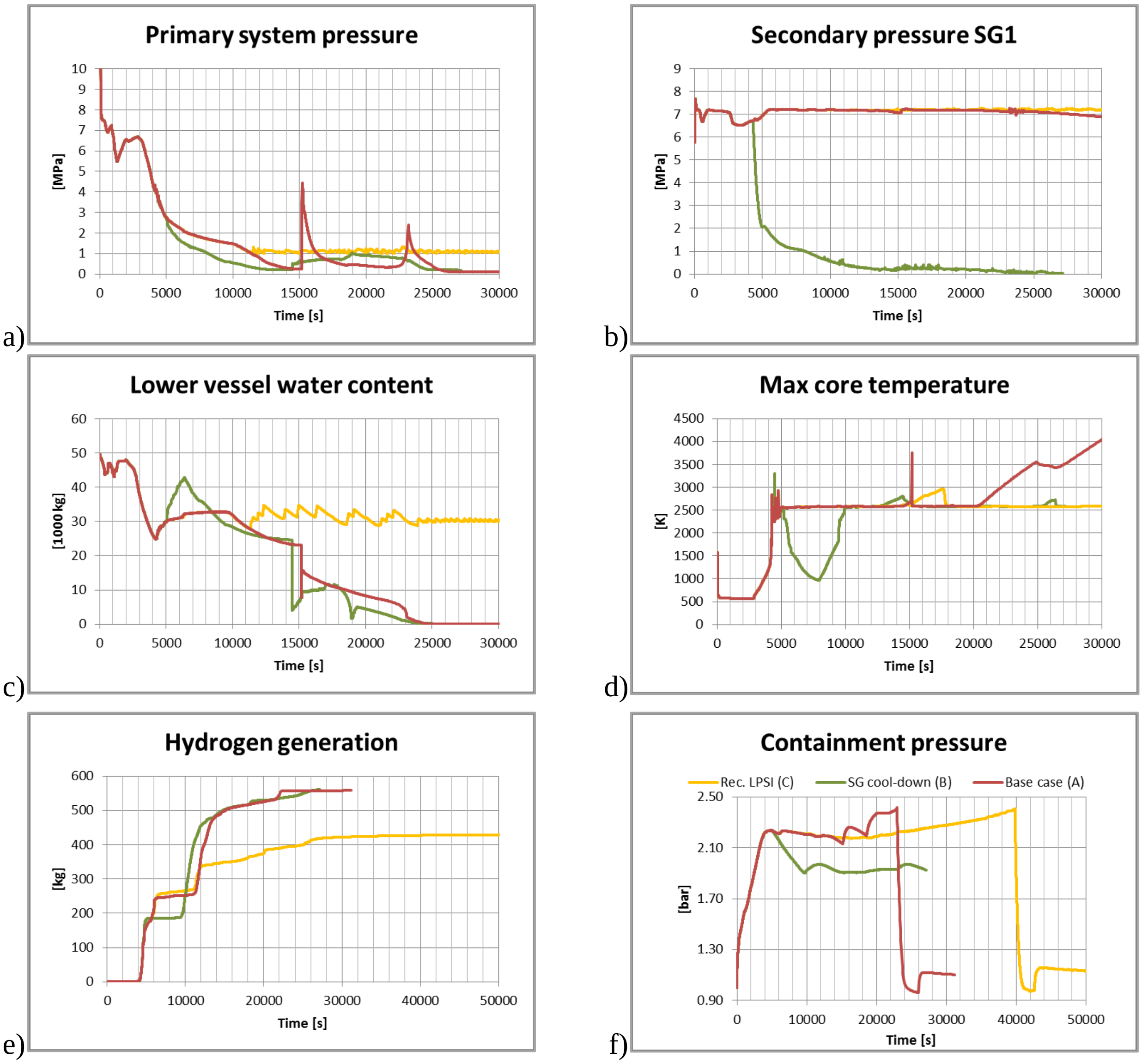

Figure 5: ASTEC simulation results for a SBLOCA, comparing a (A) base case of failed HP and LP Safety Injection with cases of (B) maximum cool down of all steam generators and (C) recovered LPSI after 2 hrs.

Key observations are:

1. In the base case (A: red in Figure 5), the SBLOCA will lead to passive accumulator injection between 4,000 s and 9700 s, begin of core degradation at 4,200 s, relocation of the melted core into the lower plenum after 15,000 s and vessel rupture after 23,000 s.

2. In case B (green in Figure 5) a lower RPV pressure due to SG heat removal is predicted. This will increase the removal of heat as long as the passive accumulators are feeding water into the primary, which leads to a significant cooling of the still almost intact core (see Figure 5d). Hydrogen generation is stopped during this period. A core melt could be avoided in such a case, if sufficient water would be injected into the primary circuit thereafter e.g. by starting the LPSI system. The results show that the SAM action of depressurizing the SGs would be highly effective in decreasing the primary pressure below the set point of 1 MPa of the LPSI, thus allowing this system to become effective after 8,000 s (Figure 5a), i.e. creating the conditions for supplying coolant about one hour earlier than in case C. If no water is available, case B will lead to core damage with timing similar to the base case.

3. In case C (yellow in Figure 5) the start of the LPSI alone, without SG heat removal (SAM action), after the RPV pressure has fallen below the set-point does not provide sufficient core cooling. However, the injected amount of water is sufficient to retain the RPV water inventory at a constant level and limits damages to a part of the core. Hydrogen production as shown in Figure 5e is continuing beyond the depicted time interval. A prolonged calculation over 190,000 s shows that hydrogen production stops after 60,000 s and that no core relocation takes place.

4. As studied in case A and C, Figure 5f shows how effectively the containment pressurisation is stopped when the CSS automatically starts on reaching its set point of 0.24 MPa.
This comparison shows that adequate SAM action (case B) can indeed improve the situation of the havarized reactor for a certain time window, and create the conditions for supplying coolant about 1 hour earlier than without action (case C). It is expected that starting LPSI in this case would enable sufficient core cooling.

3.3.2 Conclusions

As a part of the SAM application tasks in CESAM a set of “reference” ASTEC input decks for the main generic types of NPPs in Europe (PWR, BWR, VVER) have been created. These input decks have been applied for accident management of severe accident plant analyses. In particular these input decks have been employed to conduct plant analyses and possible improvements of SAM measures based on various plant scenarios and to take in consideration the lessons drawn from the Fukushima.

The analyses performed show that ASTEC is capable to simulate SAM measures in integral plant applications taking into account all the phenomena investigated in section 1.3.2 like re-flooding of degraded core, external vessel cooling, MCCI, hydrogen deflagration in the containment and passive recombiners and fission product behaviour in filters or during pool scrubbing processes.

Model development performed during the transition from ASTEC V2.0 to V2.1 allowed the application of SAM measures like NIS PARs or the application of newly developed models to plant types beyond PWRs like BWR and PHWRs. This process has been largely supported by the model development performed within CESAM as described in section 1.3.1, but is also caused by intensive code extensions and improvements by code developer IRSN on the restructuring of ICARE geometries and CESAR/ICARE coupling. The analyses of SFP accidents have to be classified specifically. While all severe accident simulation codes are designed for application to NPP power plant reactors, they lack possibilities to model the completely different geometry of SFPs besides limited validation of air oxidation models. Therefore, these analyses can only be seen as feasibility studies, because SFP studies only gained increased attention after the Fukushima accidents and the state-of-the-art for SFP modelling is still limited in the whole severe accident community.

The current status of the ASTEC V2.1 series clearly allows performance of plant analyses with SAM measures, while the release of the latest revision V2.1rev1 has been delivered only 4 month before the end of the CESAM project. This revision has shown improved stability in comparison to earlier code versions, but limited experiences with the models of the ASTEC V2.1 code restrict most analyses to the in-vessel phase, although some ex-vessel analyses exist as well. Especially, in case of the BWR reactor, creation of an input deck from scratch and missing experiences of newly developed ICARE geometry and revised CESAR/ICARE coupling led to only limited results caused by numerical code crashes. In that respect, it is worth noting that few anomalies have been recently detected by IRSN in the BWR generic input deck, making necessary to proceed to some updates in the original dataset (revised input deck currently under testing). Anyway, despite an improved robustness brought by the latest revision V2.1rev1, the work performed shows the continued need to increase the code stability of ASTEC V2.1 version in subsequent revisions in the future. On the other hand, enhanced user experiences with the logic of the new V2.1 series are still necessary to an increased usage of ASTEC for sophisticated plant analysis. In general it has to be pointed out that the base-line goal of WP40 has been achieved by creating ASTEC V2.1 reference input decks for all relevant reactor types operated in Europe as described in the report D40.45 “Set of final ‘reference’ NPP ASTEC input decks” and performance of accident calculations using these input decks with special respect to the applicability of SAM measures as described in D40.44 “Synthesis of evaluation of the impact of SAM actions through ASTEC NPP calculations”.

Finally, it is worth noting that, as recommended by the CESAM consortium, IRSN collected just at the end of the CESAM lifetime all these ASTEC V2.1 generic NPP reference datasets. So, IRSN is now being analysing the collected files to clean and possibly improve these input decks (e.g. to remove code crashes as abovementioned and/or to respect some guidelines) before proceeding to their integration in the ASTEC code package for subsequent delivery to all users in future releases. In particular, improvements have been already applied by IRSN in the collected BWR and VVER-1000 generic input decks thus to become soon available for all ASTEC users’ club members.

4 Potential impact

The work performed assured to reach the ultimate aims of the project. These are contributions and recommendations for the enhancement of SAMs in European NPPs based on the understanding of the severe accident phenomena and especially the ones relevant to the Fukushima accidents. This has been achieved by:

- Enhancement of ASTEC models for severe accident phenomena in general and especially those relevant to the Fukushima accidents and associated validation.
- Application of the new ASTEC V2.1 to PWRs, VVER440, VVER1000, PHWR and SFP and first fully integrated applications to scenarios in BWR.
- Lessons learned from Fukushima with respect to SAM.
- Calculation of scenarios in all relevant reactor types currently in operation in Europe with regard to the applicability of existing or potential future SAM measures.

This project has significantly advanced the ASTEC code development to deal with BWR scenarios. Plant analyses based on various plant scenarios accounting for the lessons drawn from the Fukushima accidents have been performed and suggestions for possible improvements of SAM measures derived.

The CESAM project helped to maintain Europe's leading role in nuclear safety by developing the required reference computation tool for the indispensable safety assessments of the Gen. II - III NPPs and mitigation measures.

Links have been maintained with other European Framework Programmes dealing with severe accidents that are conducted in the EURATOM FP7 of the European Commission or through other platforms like the NUGENIA association. This helps to promote the European tool ASTEC in the world wide severe accident community. Thus, European expertise is gained on SAM measures and on the understanding of severe accident scenarios like those in the Fukushima plants.

4.1 The ASTEC community

One impact of the CESAM project was the support of the international ASTEC users' community with the conduction of yearly CESAM workshops. Two of the four workshops performed in CESAM life time have been conducted as ASTEC user's club meetings and were open to the whole ASTEC community. Also the last workshop was connected to a workshop on ASTEC capabilities for BWR modelling with special focus on the Fukushima Dai-ichi accidents and was open to a wider audience with the exception of the CESAM concluding session. Dissemination of project results has been done through participation in conferences and publishing in journals as well as through the open web site www.cesam-fp7.eu. Additionally, subsequent release of ASTEC newsletters has been performed through this web site. Finally, international cooperation has been supported through the set-up of a mobility programme, which allowed the performance of three mobility actions for young researchers, who spent about six months with the code developers to work on specific tasks related to validation, modelling and code application. In addition, an e-learning module was produced in the frame of CESAM by JRC Petten as part of the dissemination activities that is available under astec.jrc-interactive.eu. The e-learning tool is designed to support the use of ASTEC by providing an overview of its features; it is addressed to students and new users of the code. The code development team at IRSN contributed to set up 12 modules on issues ranging from general descriptions of a severe accident, ASTEC evolution, code features needed by users to develop plant input decks. The users can follow slides and spoken text for each module of the e-learning tool.

4.2 Final remarks

The CESAM project has improved the ASTEC code for the simulation of phenomena relevant for the Fukushima Dai-ichi accidents. Especially, the new V2.1 series addresses issues necessary for realistic modelling of BWR and PHWR by a new coupling of thermal-hydraulics module CESAR in the reactor cooling system with the core degradation module ICARE, enhanced physical models and core components and enhanced robust numerics. In addition the ASTEC capabilities for support to emergency situations have been enhanced by a coupling to environmental consequence tools and estimation of the current plant status.

The validation work performed within CESAM shows the general capabilities of ASTEC to simulate most of the relevant severe accident phenomena at the state-of-the-art. CESAM-related investigations contributed greatly to extend the validation of ASTEC versions for safety-relevant issues e.g. early core degradation phase, in-vessel corium retention, ex-vessel corium coolability, hydrogen behaviour in the containment, as well as consideration of source term mitigating phenomena like applicability of filtering and pool scrubbing. Although some areas still need further modelling improvements, such limitations also apply for all current severe accident codes. Insights into code limitations and further development needs have been derived in the validation tasks performed.

A library of reference input decks for different NPP-types under operation in Europe and for several SA-sequences has been created for the latest ASTEC V2.1 code series. These generic “reference” input decks capitalize the users’ whole experience and developers’ recommendations and should allow any user to adapt them to their own NPPs in the future. Therefore, these input decks will be adjusted to every new ASTEC version and delivered as example input with subsequent code versions to ASTEC users. Example calculations using these input decks, especially with respect to severe accident measures installed in current plants, show the abilities of ASTEC to perform such analyses.

By these steps, the final goals of the CESAM project have been reached to enhance ASTEC with respect to analysis of all relevant phenomena exhibited during the Fukushima Dai-ichi accidents and improve the capabilities to be used in support to emergency centres. These new capabilities have been successfully validated and applied in plant applications with a special focus on simulation of severe accident management measures.

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