30th NENE 2021
September 6 - 9
BLED-SLOVENIA
International Conference Nuclear Energy for New Europe

EMPLOYMENT OF THE ASTEC CODE IN THE SEVERE ACCIDENTS RESEARCH ACTIVITIES AT KIT

Fabrizio Gabrielli, Victor Hugo Sanchez-Espinoza
Karlsruhe Institute of Technology (KIT)
Hermann-von-Helmholtz-Platz, 1
76344 Eggenstein-Leopoldshafen, Germany
fabrizio.gabrielli@kit.edu, victor.sanchez@kit.edu

Onur Murat, Ahmet Kagan Mercan, Walter Tromm
Karlsruhe Institute of Technology (KIT)
Hermann-von-Helmholtz-Platz, 1
76344 Eggenstein-Leopoldshafen, Germany
onur.murat@partner.kit.edu, ahmet.mercan@partner.kit.edu, walter.tromm@kit.edu

ABSTRACT

The improvement of the performance of the modern integral codes for carrying out severe accident analyses in nuclear power plants is one of the key elements of the Nuclear Safety Research program of the Karlsruhe Institute of Technology (KIT). This activity is performed in the frame of the participation of KIT in European projects and international cooperation's on the safety assessment of nuclear power plants and also aims at supporting decision making under high uncertainties for all emergency situations. The use of the Accident Source Term Evaluation Code (ASTEC), developed by the Institut de Radioprotection et de Sûreté Nucléaire, plays a central role in the KIT strategy on the research activities of severe accidents in LWRs. In the paper examples of the most recent applications of the ASTEC code at KIT in this research field are presented.

1 INTRODUCTION

The improvement of the performance of the modern integral codes for carrying out Severe Accident (SA) analyses in Nuclear Power Plants (NPPs) is one of the milestones of the Nuclear Safety Research program (NUSAFE) of the Karlsruhe Institute of Technology (KIT). The NUSAFE's mission is in fact the expertise preservation to assess the safety of any reactor system having also in mind that different NPPs operating and going to be built in EU and worldwide.

Such research activity is performed in the frame of the participation of KIT in European projects and international cooperation's on the safety assessment of NPPs including innovative designs, e.g. SMRs, and also aims at supporting decision making under high uncertainties for all emergency situations.

Having in mind that the safety demonstration on NPPs is mainly based on numerical tools and reference experiments, the KIT strategy on the employment of integral codes for SA research is based on clustering the computational issues (development and modelling) with the free access to the in-situ SA large infrastructures, e.g. QUENCH, LIVE, HYKA.

The KIT strategy on SA research is therefore focussed on supporting the integral codes development through Verification & Validation studies, the evaluation of the radiological

Source Term (ST) in SA scenarios, application of Uncertainty Quantification (UQ) methods to SA codes, applications of SA codes for severe accident managements assessment, and the use of High Performance computing (HPC).

In this framework, the use of the Accident Source Term Evaluation Code (ASTEC), developed by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN, France) [1], plays an outstanding role at KIT. ASTEC is nowadays the reference European SA integral code able to analyze the complete accident scenario from the initiating event until radioactive release out of the containment in Gen. II and Gen III water-cooled reactors. The code is extensively employed in the KIT strategy with respect to both the activities relevant to the analysis of SA scenarios for different NPPs but as well to the ASTEC development, as KIT officially supports IRSN in this task since 2019.

In this paper an overview of the use of the ASTEC code in the most recent SA research activities at KIT is shown. In particular, the results concerning the code development against the QUENCH-20 BWR test [2] performed at the KIT QUENCH test facility will be described. Concerning the application to SA scenarios in NPPs, the ST results for generic KONVOI and VVER-1000 NPPs will be shown as well as their use in view of the KIT activities on UQ of SA codes and the evaluation of the radiological consequences of such abnormal events.

2 THE ASTEC CODE

The ASTEC code [1] structure is modular, each of its modules simulating a reactor zone or a set of physical phenomena (Figure 1). The flexibility of ASTEC allows a quite large range of applications of the code, ranging from pre- and post analyses of experiments, e.g. QUENCH, PHEBUS, up to full plant analyses and Spent Fuel Pools (SFPs). The code is continuosly under improvement also thanks to the activities performed in the frame of the NUGENIA/SARNET Technical Area 2 AStec COMunity (ASCOM) project (2018-2022) [3], which aims at further augment the performance of the code for SA analyses and accident management by strengthening the activities of the ASTEC users community in a consistent way.

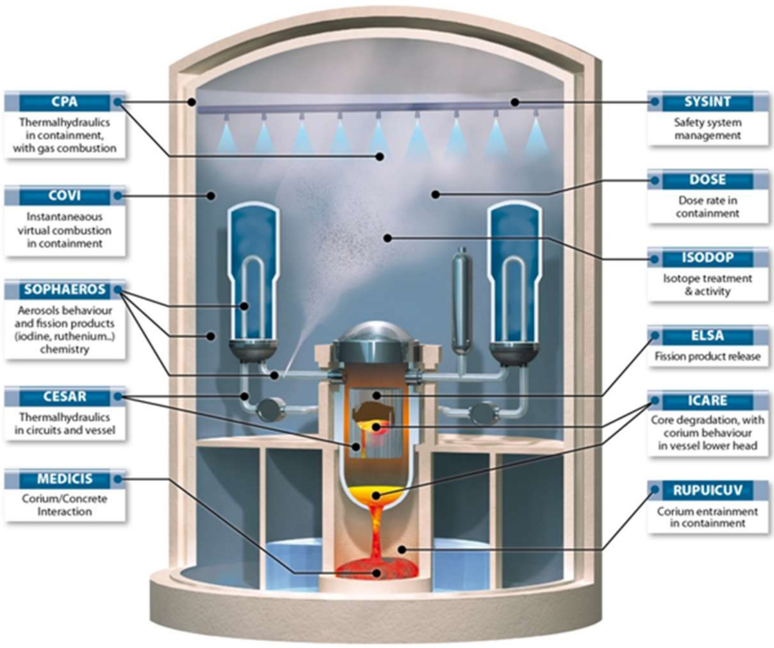

Figure 1: Scheme of the ASTEC modules [1].

In the frame of the KIT strategy on SA, the capability of ASTEC to simulate the full scenario is deeply employed, namely the simulation of the key in-vessel and ex-vessel phenomena as well as their effects on the transport of the Fission Products released by the core to the plant and the environment. In particular, the peculiarity of the ASTEC code to be able to evaluate the element- and isotope-wise composition of the radiological release is of relevance in view of the assessment of a Source Term database to be employed by predictive tools.

3 MOST RECENT APPLICATIONS OF ASTEC AT KIT

The research activities at KIT related to the use of ASTEC cover a quite large range of applications. The pre- and post-analysis of the experiments performed at KIT, e.g QUENCH, by means of ASTEC is a rather important activity in view of supporting the code development as well as the demonstration of its capabilities. In particular, pre-test analyses may also provide a support to the experimentalist, while assessing new tests.

The ASTEC analyses of full transient scenarios for different NPPs play a key role at KIT, having in mind the NUSAFE program mission. The focus is the evaluation of the radiological Source Term with two main goals: the assessment of a database of SAs ASTEC results to be analyzed by UQ methods and the evaluation of the radiological consequences of SAs to support the emergency and prepardeness teams.

The application of UQ methods of the results of SA integral codes plays nowdays an important role in the SA research community. Having this in mind, KIT participates to the on-going EU Management and Uncertainties of Severe Accidents (MUSA) Project (2019-2023) [4], with the main objective to the assessment of the capability of SA codes when modelling SA scenarios for Gen. II/III/III+ reactor designs and SFPs by using the UQ methods. With this respect massive ASTEC simulations and the developement of in-house UQ tools, i.e. the Fast Source Term Code (FSTC) [5], are performed at KIT.

The prediction of the radiological consequences of SAs are performed employing the ASTEC code in conjunction with the KIT JRODOS tool [6]. This activity, first-of-kind, allows further widening both the use of the ASTEC code and the support of KIT research in the SA field to the community.

In the next, some examples of the use of the ASTEC code at KIT will be provided.

3.1 ASTEC Validation of the QUENCH-20 Experiment

The QUENCH-20 experiment with BWR geometry simulation bundle was successfully conducted at KIT on 9th October 2019, with the objective to investigate the degradation of a BWR fuel assembly including a B_4C control blade. The test bundle is a representation of one quarter of a BWR fuel bundle of type SVEA-96 Optima-2 including the part of the blade, the water channel box and the canister to study their oxidation, and degradation under quenching conditions. The horizontal cut of the test section and the test conduct are shown in Figure 2.

The bundle consists of 24 electrically heated rods, which are approximately 2.5 m long and one corner rod, the cladding material around fuel rod simulators being made of Zircaloy-2 alloy. As shown in Figure 2, half of a typical BWR absorber cross is represented by two stainless steel blades, which includes neutron absorber material B_4C, located between fuel, the channel box and the shroud. As described in [2], the test consists of three phases. During the pre-oxidation phase the bundle is heated up to about 1250 K in an Argon and superheated steam flows of 3 g/s. In the next transient phase, the power is increased up to 18.2 kW with the same coolant flow rates. Finally, in the quench phase water at room temperature is injected in the bundle with a flow rate of 50 g/s.

The ASTEC (v2.2b) model of the QUENCH-20 test has been assessed, the radial and axial nodalization being shown in Figure 3 [7]. The models available in the code to simulate the main physical phenomena occurring in the experiment have been employed: heat exchanges (conductive, radiative, and convective), chemical interactions, oxidation, and mechanical integrity. Boundary conditions have been fixed according to the experimental data released from the KIT QUENCH team.

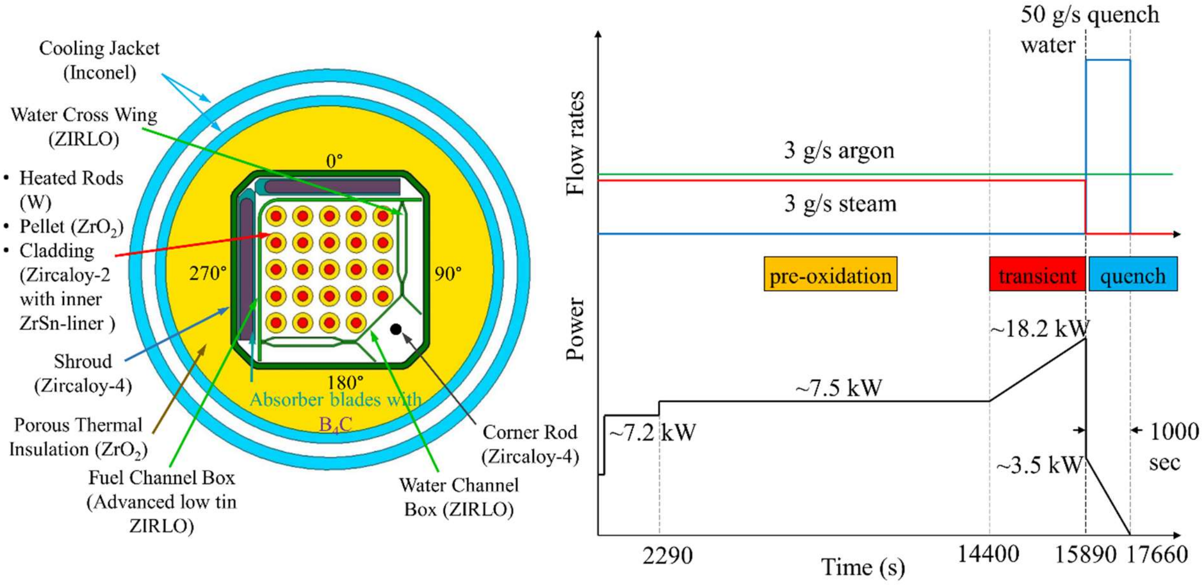

Figure 2: Horizontal cut of the QUENCH-20 test section and test conduct [2].

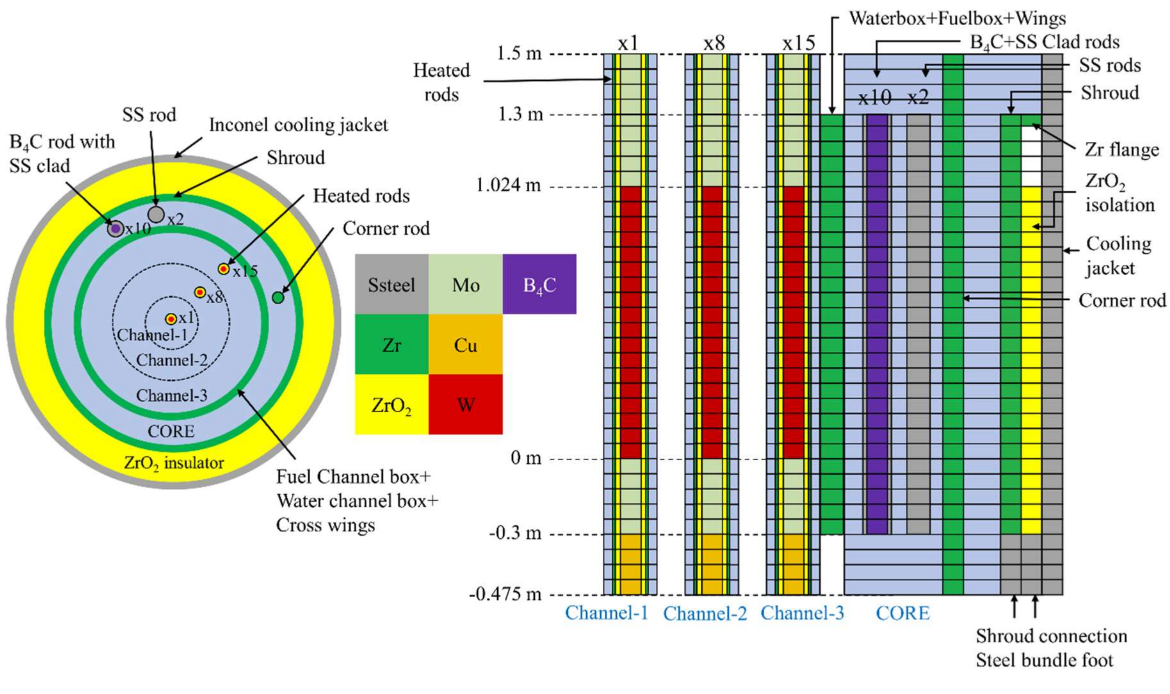

Figure 3: ASTEC model of the QUENCH-20 test: radial and axial nodalization [7].

The ASTEC and experimental results concerning the total amount of hydrogen generated during the test are shown in Figure 4. More details about the ASTEC code validation for the test may be found in [7]. The ASTEC results in Figure 4 show a rather good agreement with the experimental data during the test.

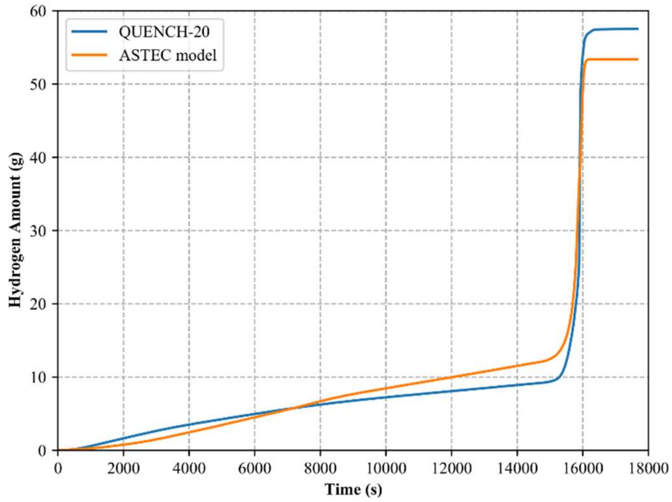

Figure 4: H_2 generation in the QUENCH-20 test: ASTEC vs. experiment [2,7].

The results concerning the total H2 generated as well as the total mass of H2 produced by the B4C oxidation are shown in Table 1, the ASTEC results showing a very good agreement with the experimental data.

Table 1: H_2 generation in the QUENCH-20 test: ASTEC vs. experiment [2, 7].

Quantity	Experiment	ASTEC
Total H_2 (g)	57.4	53.4
H_2 from B4C Oxidation (g)	10	9.48

3.2 Source Term Evaluation for Severe Accidents in NPPs

The analysis of different SA scenarios in a generic KONVOI NPP is described in this chapter. The original ASTEC input deck developed in the framework of the CESAM project and available in the code release has been further improved at KIT [8]. In particular, all the calculation modules available in the code have been activated and the related modelling has been assessed in view of the ST evaluation. The input deck is therefore able to simulate the main in-vessel and ex-vessel phenomena as well as the Fission product (FP) transport from the vessel to the environment.

In the current analyses, performed by employing the ASTEC v2.2b version, a Medium Break Loss Of Coolant Accident (MBLOCA, break diameter of 12”) in conjunction of a Station Black Out (SBO) scenario has been considered. In order to improve the level of the confidence of the ST evaluations, realistic fuel inventories have been computed at the equilibrium cycle and employed in the ASTEC modelling. The current analysis aimed at showing the effect of employing different amounts of FPs in the core on the ST. Having this in mind, two fuel inventories have ben considered: at the beginning and at the end of the equilibrium cycle, labelled as BEOC and EOEC in the next.

In the MBLOCA simulations, we assume that after the scram signal following the break (at t=0 s), the turbine admission and the main feed water pumps are closed. The conditions for the activation of the Emergency Core Cooling System (ECCS) are fulfilled, the Main Coolant Pumps (MCPs) are coasted down, and the pressure regulation in the pressurizer is switched off. The High and Low Pressure Injection Systems are activated, when the gas temperature in the primary circuit exceeds 650 °C, until the water tanks are empty. In such conditions, the core melt begins. In case of SBO, no safety systems are assumed to be available. Furthermore, when the radial erosion of the cavity is larger than 4.4 m, water ingression from the sump to this region is assumed.

The instants at which the main phenomena characterizing the progression of the accidents up to the basemat rupture are shown in Table 2. One may observe a faster core degradation at EOEC than at BEOC conditions, namely when larger amount of FPs are loaded in the core. Such a behaviour is because of the larger decay heat power produced in the core at EOEC in the first part of the accident. In such conditions the FP release from the fuel as well as the first material slump to the LP occurs much earlier. Once most of the FPs are released from the core, the accident progression is mainly triggered by the shutdown of the safety system, namely in the scenarios where SBO occurs. As a result, the failure of the Lower pressure Vessel (LPV) occurs earlier when the SA scenarios triggered by SBO are considered.

Table 2: Quicklook of the progression of the accidents.

	BEOC		EOEC	
Phenomenon	MBLOCA	MBLOCA+SBO	MBLOCA	MBLOCA+SBO
FPs Release (s)	764	764	484	514
First slump with FPs in the LP (s)	7594	1344	794	784
20/50 tons relocated to the LP (s)	9484/9494	1454/2094	13304/16214	854/1244
70/90 tons relocated to the LP (s)		2194/-	16224/16244	
LPV Failure (s)	15864	4960	19892	2801
Basemate Rupt. (s)	255512	248836	222732	168747
Total H_2 In-vessel/Containm. (kg)	782/2248	896/2193	759/2339	793/2377

The behaviour of the containment pressure in all the sequences is shown in Figure 5.

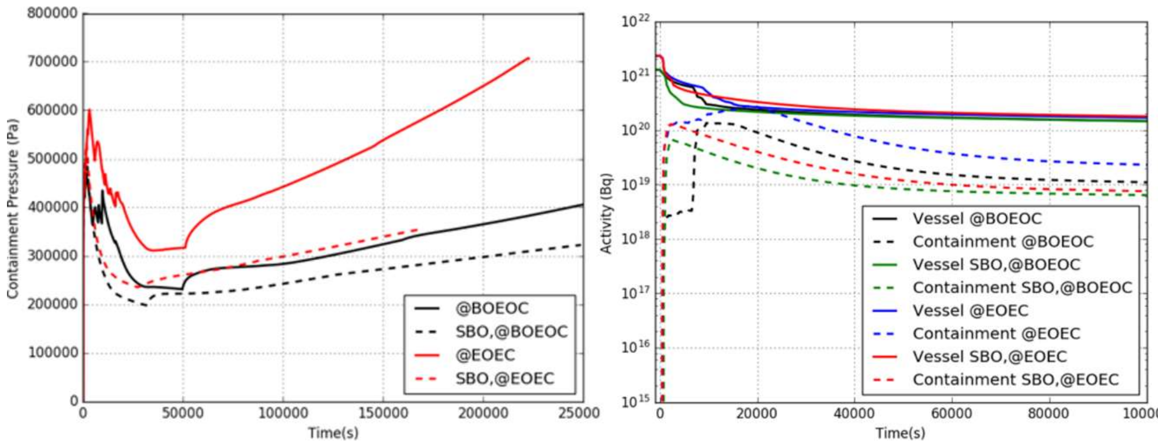

Figure 5: Containment pressure and activity in the vessel and in the containment.

The results show that the ASTEC simulations for the MBLOCA at the EOEC scenario lead to much higher values of the pressure also triggered by the hydrogen produced in the cavity due to water flooding. In such scenario, the pressure reaches about 7 MPa at the basemat rupture instant. The much faster core degradation in the SBO scenarios show a first fast peak (about 5 MPa), followed by a trend 'milder' than in the MBLOCA cases.

In view of the ST evaluation, the results of total activity in the vessel and in the containment is shown in Figure 5. One may observe that the largest part of the total activity in the vessel is transported to the containment in the MBLOCA (EOEC) scenario (dashed blue line).

The Xe and Cs aerosol in the containment are shown in Figure 6 as fraction of their initial core loading. The results for the MBLOCA scenarios at BOEC and at EOEC look rather similar, despite different in amplitude. Concerning Xe about 98% and 90% of the initial core loading are released to the containment. Furthermore, the maximum amount of Cs aerosol reaches about 18% of the initial loading at EOEC, while it is about 12% at BEOC. The results also show a much lower amount of Cs aerosol transported to the containment in the SBO scenarios.

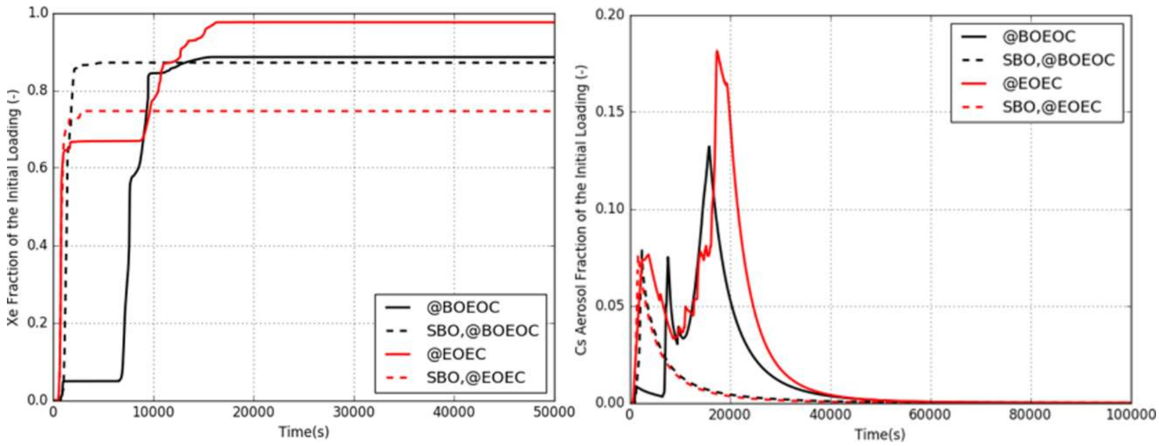

Figure 6: Xe and Cs aerosol in the containment.

As mentioned above, UQ of the ST results of SA scenarios in a generic KONVOI NPP is currently going on in the frame of the participation of KIT to the EU MUSA project. With this goal, large calculation campaigns have been performed aiming at building an enough large database for the application of UQ methods. Such database is also a prerequisite for the performing ST prediction analyses in the frame of the KIT and Framatome GmbH joint collaboration in the German WAME project. WAME, funded by the German Federal Ministry of Economic Affairs and Energy, aims at the development of a novel real-time program system to improve decision making in SA events in nuclear power plants [9].

3.3 Radiological Dispersion in the Environment

A calculation route to evaluate the radiological consequences of SAs in NPPs has been assessed at KIT by employing the ASTEC and the JRODOS tool. Such first-of-kind simulations have been performed at KIT for a Large Break Loss of Coolant Accident (LBLOCA) with SBO scenario in a generic VVER-1000 NPP. Note that VVER NPPs are currently operating in the Eastern Europe and four units (type VVER-1200, AES 2006) are under construction in Turkey.

An ASTEC (v2.2b) input deck has been developed at KIT in order to perform full SA scenario analyses [10]. Efforts have been performed to significantly improve the input deck available in the code release. Having this in mind, detailed modelling of key components of the plant, i.e the horizontal steam generator, has been carried out as well as the activation of all the ASTEC calculation modules to simulate the vessel to the environment FP transport.

The ASTEC results of the cumulative H_2 production during the LBOCA with SBO scenario are shown in Figure 7. The results show that about 200 kg are produced during the in-vessel phase and about 800 kg are produced due to the MCCI phenomena in the cavity.

In order to evaluate the radiological dispersion in the environment, the ASTEC results of the activity of the main FPs released from the NPP are evaluated and provided to JRODOS. The results for a selected list of FPs are shown in Figure 7 (right side).

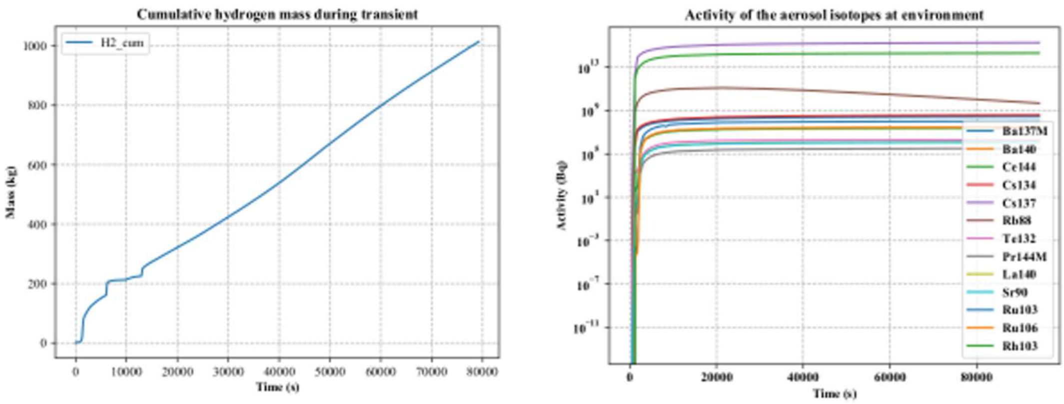

Figure 7: H_2 production in a LBLOCA+SBO scenario in a generic VVER-1000 NPP (left) and activity (Bq) of selected FPs released to the environment (right).

Preliminary ASTEC/JRODOS simulations have been performed by assuming winter weather conditions (data for 23 hours used) and by employing the RIMPUFF atmospheric dispersion model. The NPP is located at KIT and the JRODOS calculation range is 800 km. Finally, no emergency preparedness plans have been used.

The results of the JRODOS simulations concerning the dry and wet ground contamination and the total potential dose rate are shown in Figure 8. The results show that the ground contamination is dominated by the Cs^{137} and Ba^{137M}, the contributions to the total being 97% and 2%, respectively. The maximum activity density for such FPs is 1.24 MBq/m2 (Cs^{137}) and 19.8 kBq/m^2 (Ba^{137M}). The total effective dose reaches 3.32 mSv/hr around the accident site and decreases to 1.0 microSv/hr at 100 km. The results also show that Cs^{137} dominates (about 98% of the total) the dose rate to the bone marrow (3.18 mSv), lungs (3.11 mSv), and thyroid (3.19 mSv).

The ASTEC/JRODOS calculation route has been therefore successfully assessed for the first time. The preliminary results provide a solid basis of understanding in view of the application of the methodology to realistic analyses for any NPP located in the world, taking into account the large weather database available in JRODOS.

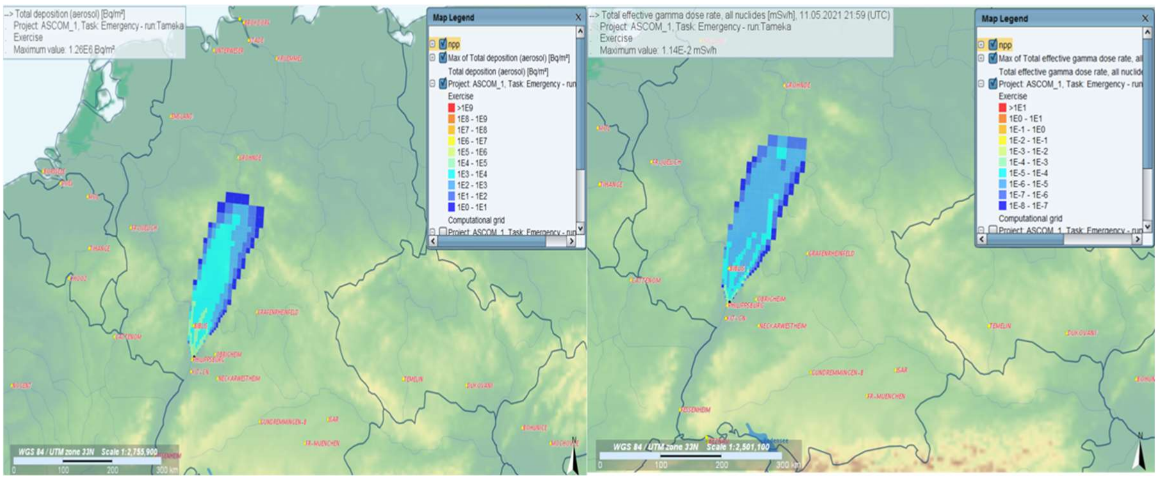

Figure 8: JRODOS results: ground contamination and total potential dose rate.

4 CONCLUSIONS AND OUTLOOK

The European reference ASTEC code, developed by IRSN, is extensively employed in the SA research activities performed at KIT in the framework of the NUSAFE program. Several SA-related researches are exploited, ranging from the V&V of integral codes to the analysis of SA scenarios in different types of NPPs, with main focus on ST evaluation. Furthermore, efforts are spent to support the ASTEC development, as KIT officially support such activity since 2019.

The possibility to access to large SA infrastructures, makes KIT a rather unique campus where experimental data may be employed for the code validation. With this respect, the analysis of the QUENCH-20 BWR test has been described, the ASTEC results showing a very good agreement with the experimental results.

The results of the evaluation of the ST in SA scenarios in a generic KONVOI and VVER-1000 plant have been shown. Additional to the typical analysis of the scenarios, the ASTEC code is employed for the assessment of database of SA results in view of the application of UQ methods to both identify the effect of the existing uncertainties of the ASTEC physical modelling on the ST predictions. Furthermore, ASTEC is also used to compute the necessary data to evaluate the radiological impact of SA scenarios by means of the KIT JRODOS tool. Such first-of-kind activity is of relevance to support the emergency and preparedness in case of SA events.

A solid basis has been therefore assessed in the different aspects of the KIT SA strategy in line with the necessity to maintain the expertise preservation of the nuclear reactor safety of different NPPs both in operation and going to be built in EU and worldwide. Having in mind this mission, ASTEC is planned to be employed at KIT in research activities on innovative systems. As an example, KIT is involved in the OECD/NEA and IAEA international activities on the development and testing of Accident Tolerant Fuels (ATFs), which are candidated to be employed in the Generation IV reactors, both in the KIT QUENCH test facility as well as in the ASTEC modelling of such innovative components.

REFERENCES

[1] P. Chatelard, at al., "Main modelling features of the ASTEC V2.1 major version", Ann. Nucl. Energy 93, 83–93. https://doi.org/10.1016/j.anucene.2015.12.026.

[2] J. Stuckert, et al., "First results of the QUENCH-20 test with BWR bundle", 25th International QUENCH Workshop, Karlsruhe, Germany, Karlsruhe Institute of Technology, 22-24 October 2019, pp. 106-127. https://doi.org/10.5445/IR/1000100298.

[3] NUGENIA/SARNET ASCOM Project, https://snetp.eu/portfolio-items/ascom/.

[4] L.E. Herranz, at al., "The EC MUSA Project on Management and Uncertainty of Severe Accidents: Main Pillars and Status," Energies, 14 (2021).

[5] A. Stakhanova, et al., "Uncertainty and sensitivity analysis of the QUENCH-08 experiment using the FSTC tool", 2021, submitted to Annals of Nuclear Energy.

[6] W. Raskob, et al., "JRODOS: Platform for improved long term countermeasures modelling and management", Radioprotection, vol. 46, n° 6 (2011) S731-S736.

[7] O. Murat, at al., "Preliminary validation of ASTEC V2.2.b with the QUENCH-20 BWR bundle experiment", Nuclear Engineering and Design, 370, 2020, 110931. https://doi.org/10.1016/j.nucengdes.2020.110931.

[8] F. Gabrielli, SARNET-ASCOM-RUNPLANT-R26, 2021.

[9] WAME Project, German Federal Ministry of Economic Affairs and Energy, funding code FZK 1501582.

[10] A. K. Mercan, "ASCOM Technical Report SARNET ASCOM RUNPLANT R7," NUGENIA, Karlsruhe, 2021.