Kit Activities and Infrastructure for Education and Training on Knowledge Management in Reactor Safety and Technology

1V. H. Sanchez-Espinoza, W. Tromm, R. Stieglitz
Karlsruhe Institute of Technology (KIT), Hermann-von-Helmholtz-Platz 1, D-76344 Eggenstein-Leopoldshafen, Germany

Abstract
Karlsruhe Institute of Technology (KIT) is a research university within the Helmholtz Association (HGF) with about 9,500 employees, and 25,000 students. It bundles the missions of a university of the state of Baden-Wuerttemberg and of a large-scale research institution of the Helmholtz Association. Within these missions, KIT is operating along the three strategic fields of action of research, teaching, and innovation, including energy. KIT is devoted to top research and excellent academic education as well as to being a prominent location of academic life, life-long learning, comprehensive advanced training, exchange of know-how, and sustainable innovation culture. In the energy branch, the Nuclear Waste Management, Safety and Radiation Research program (NUSAFE) has a long tradition, is widely recognized and represents an integral part of national provident research providing core competences on the internationally highest level of science and technology regarding nuclear safety and waste management research. This paper will present and discuss the activities devoted to preserve nuclear knowledge and to foster the education in key-areas important to solve the challenges in nuclear waste management, reactor safety, and radiation protection and decommissioning of nuclear power plants. In addition, the courses offered by the AREVA nuclear professional school at KIT and the activities and KIT-infrastructure consisting of unique experimental facilities will be discussed to explore possible co-operations with Latin America. Examples of developed courses (e.g. online courses) in the frame of the KIT-participation on European projects devoted to E&T e.g. GENTLE, ANNETTE will be given e.g. under development. Finally, possibilities for co-operations of KIT with Latin America universities will be outlined.

1 Introduction
The Programme NUSAFE (Nuclear Safety Nuclear Waste Management and Safety Research) is divided in two Topics: a) Nuclear Waste Management and b) Reactor Safety. The scientific activities and education and training of scientific staff are carried out in close cooperation with partners in the Alliance for Competence in Nuclear Technology and various national and international, notably European community collaborative projects, especially with universities. The NUSAFE program pools both the R&D- and the teaching activities within the Helmholtz Association (HGF) where three Helmholtz centres are involved: the research Centre Jülich (FZJ), the Helmholtz Zentrum Dresden Rossendorf (HZDR) and the Karlsruhe Institute of technology (KIT). These research centres traditionally maintain a close co-operation with the universities around their sites [1]. The Topic “Nuclear Waste Management” deals with basic and applied research related to the scientific basis for the safe disposal of nuclear waste in a deep geological repository and provides a (radio-) geochemical contribution to site selection and characterization. Since the safety of a repository system over several hundred thousands of years cannot be demonstrated by technical means only, a fundamental understanding and quantification of the chemical, biological, and physical processes of radionuclide behaviour are essential for safety assessment.

1 E-mail of first author: victor.sanchez@kit.edu
In addition, the study of innovative waste management strategies with respect to the decommissioning of nuclear facilities and (historic) legacy waste in Germany is a scientific goal, too. Last but not least, the scientific and technological basis for the partitioning and transmutation strategy is being explored to provide sufficient scientific know-how for a comprehensive evaluation of its feasibility, advantages, and drawbacks in the European context.

The topic “Reactor Safety” is focused on research for the safe operation of nuclear power plants in Germany and abroad as well as design-basis and beyond-design basis accidents in nuclear power plants or nuclear facilities. Doing so, Germany will maintain its influence in international safety organisations and support management and independent evaluation of severe nuclear accidents like the one in Fukushima. A key pillar is the further development and validation of numerical simulation tools using data of KIT-facilities or from abroad dedicated to safety-relevant phenomena expected to occur during accidents.

This paper describes the KIT infrastructure and the scientific activities related to numerical simulation tools for the evaluation of design and beyond design basis accidents (DBA) of Light Water Reactors (LWR) highlighting their relevance for education and training activities of students at university level but also of doctoral students and young scientists and researcher of different stake-holders e.g. regulators, TSO, manufactures, utilities, international organizations dealing with reactor safety.

2 The KIT Research for design basis accidents

The KIT scientific objectives are to continuously develop and improve the safety analysis tools to allow for a state-of-the-art safety assessment of NPPs. This covers also aspects arising from the German phase-out decision, e.g. aspects of the post-operational phase and issues related to the interim storage of fuel elements. The development of tools for the safety analysis of reactor concepts being under development in neighbouring countries is an important issue. This includes thermal-hydraulic investigations of different coolants, neutron-kinetics, thermo-mechanics and safety aspects of reactor concepts of different generations [2]. Safety analyses are mainly based on both deterministic and probabilistic methodologies in combination with experimental validation performed by partly unique large scale experimental facilities. Code validation is strongly connected with experimental work providing the corresponding experimental data of high-quality and high resolution.

The KIT approach combines numerical simulations and experimental investigations of safety-related phenomena in relevant areas of reactor safety [3]. Hereafter, selected experimental facilities of KIT infrastructure for LWR and innovative reactors will be shortly presented in the first part. It will be complemented by the numerical activities going on in this area.

2.1 Experimental investigations for design basis accidents

One of the peculiarities of HGF research centres is the capability to operate “large experimental facilities” and to perform safety-related investigations in the reactor safety area. Selected facilities currently under operation or commissioning will be described including the main research goals.

2.1.1 The COSMOS-H facility

This COSMOS (Critical Heat Flux On Smooth and MOdified Surfaces-High Pressure Loop) is a facility being built for the experimental investigation of boiling- and other complex flow phenomena occurring in nuclear power plants [4] and [5]. After commissioning of the facility, these physical phenomena are studied under prototypic thermodynamic conditions with
pressures up to 17 MPa and temperatures up to 360°C. The facility permits a heating power of 2 MW for heating and evaporation, see Figure 1. It uses liquid water and steam as working fluid. The special quality of the experimental results is assured by the unique instrumentation of the water loop and test section in combination with high pressures and temperatures. Measurements for the development and validation of new CFD methods will be performed. The objective is the visualization and recording of flow conditions at the onset of boiling crisis under reactor typical conditions. Firstly, the test section will be equipped with a single heated rod in an annular gap but in future it is planned to perform measurements at a rod bundle. The commissioning of the COSMOS-H is foreseen for 2018. In the meantime, series of tests were carried out in the low-pressure loop COSMOS-L (low pressure) enhancing the database on CHF for different mass flow rates, inlet temperatures and pressures [5].

Figure 1 COSMOS-H Facility for CHF investigations

2.1.2 The QUENCH-LOCA test program

The QUENCH facility is the successor of the CORA facility devoted to investigate the behaviour of overheated and partly oxidized and degraded fuel rod bundles of LWR quenched by cold water or steam. More than a dozen of tests for fuel rods of different reactor types were performed to understand the degradation phenomena during the early in-vessel phase of severe accidents [6]. The LOCA is one of the major accidents that is investigated and for which acceptance criteria are defined by the regulators of different countries e.g. regarding the maximal cladding temperature, the cladding oxidation degree to express the cladding degradation due to embrittlement and hydrogen uptake. Currently, these criteria are being revised and for it new experimental investigations are needed that allow a more precise understanding of the degradation phenomena of different cladding materials under e.g. high-burn-up conditions.
For example, new experimental findings have shown that hydrogen uptake especially during ballooning and burst of the fuel rod -secondary hydrogen uptake- strongly affects the mechanical properties. Hence, the QUENCH-LOCA program has been started at KIT in cooperation with and supported by German utilities to investigate these phenomena in large-scale bundle experiments and to generate detailed data for code validation purposes [7]. The aim of such investigations was the development of new embrittlement criteria for the German nuclear reactors. The KIT QUENCH test bundle widely used for the analysis of the early-phase of severe accidents in LWR was selected to perform the QUENCH-LOCA bundle test series, see Figure 2. Compared to single-rod experiments, bundle tests have the advantage of studying the mutual interference of rod ballooning among fuel rod simulators as well as the local coolant channel blockages in a more realistic arrangement. The program was started in 2010 with the QUENCH-L0 scoping test used 21 heated rods with as-received Zry-4 claddings [8]. Then advanced claddings e.g. Duplex D4, M5®, and ZIRLO™ were selected for the experimental investigations. The QUENCH team is involved in the supervision of master and doctoral students as well as hosting foreign visitors for research stages in the frame of different co-operations.

![Figure 2](image.png)

*Figure 2* The KIT QUENCH test facility with the test section (left) and the radial cut though the test section showing the location of the fuel rod simulators [7]

It is worth to note that the QUENCH team is hosting since more than two decades the international QUENCH workshop at KIT where different experts from all around the world discuss the latest results of the experimental investigations and the validation and application of severe accident codes. More information can be found in: [http://quench.forschung.kit.edu/250.php](http://quench.forschung.kit.edu/250.php).

2.1.3 The KASOLA facility

The experimental sodium loop KASOLA (KArlsruhe SODium LABoratory) is a facility characterized by high flexibility with respect to a wide spectrum of thermal hydraulic experiments that can be done. It includes a sodium inventory of 7 m³ and it can operate up to
550 °C. A magneto-hydrodynamic (MHD) pump can deliver a flow rate of around 150 m³/h at a pressure head of 0.4 MPa. Unique tests such as flow direction reversal and numerous hypothetical scenarios can be investigated thanks to the peculiar features of this type of pump [9]. The base loop consists of two test ports, a versatile test section and a pool test section. The versatile test section has a maximum effective height of up to 6 m to be used for component validation tests at almost typical axial heights, as well as generic thermal-hydraulic benchmark experiments. In the pool section, which has the dimensions 4x4x0.4 m³, a core simulator, an intermediate heat exchanger, and decay heat removal coolers can be hosted to simulate the flow in a sodium pool nuclear plant, see Figure 3.

Figure 3 The KASOLA facility with the main components [9]

The flow patterns in the sodium pool of a nuclear plant will be studied in detail on the basis of a Hele-Shaw approximation, for the transition between forced, mixed, and natural convection. In addition, measurement techniques can be qualified and in-service inspection and repair (ISI&R) measures for sodium cooled reactor systems can be tested. On the other hand, the auxiliary port will be used for smaller experiments. Due to the KASOLA-flexibility not only nuclear applications but also aspects of thermo-electric energy conversion, such as the study of Alkali Metal Thermal Energy Converter (AMTEC), as well as thermal storage using liquid metals can experimentally investigated in this facility [10].

2.1.4 The KALLA Facilities
In order to support safety-related investigations of innovative systems e.g. MYRRHA, ALFREWD, ASTRID, SEALER cooled by liquid metals e.g. lead, lead-bismuth and sodium, experimental facilities are build and operated at different places in Europe such as in TALL in Sweden, CIRCE in Italy and KALLA, DRESSDYN, and KASOLA in Germany [11] and [12]. To build and run MYRRHA and ASTRID reliable data and experiences regarding instrumentation and operation of liquid metal facilities, understanding of related turbulent
heat transfer processes derived from standard configurations such as an annulus are urgently needed. For example, the safety assessment of the MYRRHA facility requires deep understanding of coolability of the core fuel assemblies in both normal and postulated accidental conditions. The Karlsruhe Liquid Metal Laboratory KALLA covers experimental investigations of thermal-hydraulic and core components; steam generator and cooling safety; coolant chemistry control and HLM corrosion; fuel and fuel safety, etc. To support the phase-out the KALLA has also reoriented and refocused its research activities on three major fields of liquid-metal (LM) technology: a) Experiments for cooling of core components (rod cooling and spallation target thermal hydraulics) of Accelerator Driven Systems (ADS) b) Experiments and modeling of direct pyrolysis of methane in a liquid metal bubble column reactor and c) Fundamental numerical studies and concept development of a solar furnace for the investigation of LMs as efficient heat transfer fluids for concentrated solar power (CSP) systems [13]. One of the KALLA tests is the rod bundle test dedicated to the study of heat transfer of lead–bismuth eutectic (LBE) cooled rod bundles, under conditions representatives of the reactor, in different configurations. This includes the effects of the selected spacer concept (grids or wires) on the temperature distribution, as well as the mechanisms leading to flow blockages and their consequences [14] and [15], see Figure 4 and Figure 5. The results are expected to fulfil two complementary objectives. First, the safety assessment of the core components in a postulated thermal-hydraulic scenario is supported due to the prototypical features of the experiments. Second, reliable validation data is provided for predicting models of sub-channel, CFD and system codes, essential for extrapolating the results, filling the unavoidable gap between the experimental setup and actual reactor application. The KALLA is also involved in education, training and dissemination activities as part of the different European projects for which tests are being performed and hence the KALLA team is hosting international workshops and also visiting students, advising master and doctoral thesis in the relevant fields.

Figure 4 Schematic view of the KALLA facility with the location for the LBE rod bundle experiment

Figure 5 The KALLA test bundle
2.2 *The numerical simulation tools for LWR design basis accidents (DBA)*

Numerical simulation tools are being used by regulators and technical support organizations for safety demonstration and hence they must reflect the state-of-the-art in science and technology. Therefore these simulation tools are under continuous improvements to account for core design changes e.g. heterogeneous core loading, increased burn-ups. For long time, best-estimate thermal hydraulic codes with point or 1D kinetics have been used in the nuclear industry and by regulators. Nowadays, the development of 3D neutron kinetics models based on the neutron diffusion approximation is mature. There, the core is subdivided in many 3D nodes (e.g. 20 axial nodes and 1 radial node per fuel assembly) and the influence of the thermal hydraulic on the neutronics is taken into account via homogenized and condensed cross sections. The quality of the core simulator predictions heavily depends on the quality of the generated nodal cross sections. The accurate simulation of modern LWR cores requires the introduction of corrections for the nodal cross sections to take into account the heterogeneities and the larger gradients between e.g. a UOX and MOX fuel assembly, between a FA with control rods and another without or between different FA-types and the radial reflectors. Hence the development of reactor dynamic codes with higher order transport solutions instead of nodal diffusions solvers is being pursued in different places worldwide [16], [17] and [18]. To increase the prediction accuracy of important phenomena such as the coolant behaviour within the core and reactor pressure vessel in case of transients with pronounced local power distortions, the inclusion of 3D thermal hydraulic, subchannel or CFD codes instead of 1D parallel channels system thermal hydraulic codes is required. The in-house code development is concentrated on one side on thermal hydraulic subchannel and porous media two-phase flow codes and on the other hand on advanced multi-physics and multi-scale coupling approaches. Consequently, KIT cooperates over 20 years with the US NRC in the frame of the international CAMP-program together with national key-players are industry, regulators, universities and R&D in the validation and application of safety assessment tools in the field of reactor dynamics, system thermal hydraulics, and uncertainty quantifications. For the improved description of the thermal hydraulic phenomena inside the RPV and the core, KIT is developing a subchannel code, SUBCHANFLOW [19], and a porous media two-phase flow code, TwoPorFlow [20], to be used as stand-alone or coupled with thermal hydraulic system codes or with neutron kinetics solvers. The aim of the SUBCHANFLOW development is to have a fast running simulation tool for both LWR and liquid and gas cooled reactor systems able to perform full core pin-wise simulations while the TwoPorFlow development [21] and [22] aims to fill the gap between CFD and system codes. Validation work is performed using relevant data for Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR) [23]. To take into account the cross flow between fuel assemblies (FA) in case of FA-based simulations or between the subchannels in case of pin-based simulations, the KIT SUBCHANFLOW code was implemented in the NURESIM simulation platform and there it was coupled with different solvers such as COBAYA3 [24], DYN3D and CRONOS [25]. In Figure 6, it can be seen a typical online visualization of the CRONOS/SUBCHANFLOW code running inside the SALOME platform. This platform is also a very useful tool for education and training purpose of students due to its user friendly GUI-based post- and pre-processor.
Due to the limitations of legacy nodal core simulators based on diffusion approximation of codes such as CRONOS, PARCS, DYN3D, NESTLE, etc. to predict local safety parameters at pin level, high order solutions with pin-based spatial discretization are needed for both neutronics and thermal hydraulics. In addition, high-fidelity reference solutions are needed at pin-level since experimental data in such detail are very scarce. Hence, at KIT different higher order solutions at pin level are being developed. First of all the coupling of subchannel codes with a transport solver (SP3) for pin-by-pin simulations of full LWR cores under stationary or transient conditions [26] and [27]. This first-of-the-kind simulation with the new high fidelity code DYNSUB5.0 (DYN3D-SP3/SUBCHANFLOW) permitted the simulation of the REA taking into account local feedbacks at pin level, Figure 7. Secondly, the coupling of Monte Carlo codes (MCNP5 and SERPENT2) with subchannel codes for high fidelity reference solutions at pin-level for different reactor designs [28] and [29], Figure 8. There pin power predicted by SERPENT/SUBCHANFLOW for the mix-loaded core is shown.

These developments paved the way for very detailed simulations of a full PWR core at pin and subchannel level. Such simulations provide reference solutions for any low-order neutronic/thermal hydraulic coupled static or dynamic simulation for which no experimental data is available.

**Figure 6** CRONOS/SUBCHANFLOW simulation of the PWR TMI-1 core steady state conditions inside the NURESIM Platform

**Figure 7** DYNSUB5 predicted power density of a HZP rod ejection accident in [W/cm³] at the time with

**Figure 8** Pin power predicted by SERPENT2 / SUBCHANFLOW for the PWR UOX/MOX core
2.3 The KIT Severe accident research

As Chernobyl and Fukushima severe accidents demonstrate, a significant release of radioactive materials into the environment can cause severe consequences both for people's health and the country's economy. Therefore, severe accidents are the focus of considerable research worldwide. KIT combines experimental investigations with the development and validation of numerical simulation tools for safety assessment in national and international co-operations.

2.3.1 KIT experimental facilities for severe accident research

The KIT severe accident research is focused on both a) experimental investigations and b) numerical simulations [6], [30] and [31]. To the experimental facilities operated by KIT, the research is focused on in- and ex-vessel phenomena as follows:

- QUENCH facility is the only operating experimental facility in EU for investigations of the early and late phases of core degradation in prototypic geometry for different reactor designs and different cladding alloys, incl. analysis of the relocation of cladding and fuel and the formation and cooling of in-core debris beds to gain information on the characteristics of the created particles. In the reporting period the QUENCH LOCA programme was finalized with experiments investigating the coolability and uptake of secondary hydrogen during ballooning and burst of cladding tubes [6].

- LIVE-3D and LIVE-2D facilities concentrate on the investigation of the evolution of the in-vessel late phase of a severe accident, including e.g. formation and growth of the in-core melt pool, characteristics of corium arrival in the lower head, and molten pool behaviour after the debris re-melting in large scale 3D geometry with emphasis on the transient behaviour. In the reporting period, the melting behaviour of two-component debris beds was studied demonstrating that in certain transient situations the heat flux towards the vessel wall may be higher than in the bounding case with a fully developed molten pool [32].

- DISCO is the only operating facility available worldwide for integral direct containment heating and fuel-coolant interaction investigations. It is designed to perform scaled experiments that simulate melt ejection from the RPV to the reactor cavity after the RPV failure under low system pressure during severe accidents in LWRs. These experiments investigate the fluid-dynamic, thermal and chemical processes during melt ejection out of a breach in the lower head of an LWR pressure vessel at pressures below 2 MPa. In the reporting period, the tests addressed the interaction of a molten jet with a shallow water pool focusing on the premixing phase and without external triggering of the steam explosion. In 2017 the DISCO facility was upgraded with a larger cavity with a transparent wall to visualize the interaction of a pressure-driven melt jet (concentrating on the premixing phase) with a deep water pool. The tests will be performed in 2018 [33] and [34].

- MOCKA is the 3D large-scale facility to study the interaction of a simulant oxide (Al₂O₃, ZrO₂, CaO) and metal melt (Fe) in a stratified configuration with different types of concrete. The resulting solidus temperature of approximately 1360 °C is sufficiently low
to prevent a formation of an initial crust at the oxide/concrete interface observed in the CCI experiments (ANL, USA). The oxide/concrete interface contact temperature in MOCKA test was estimated to be 1400 °C. To allow for a long-term interaction, internal heating was provided by alternating additions of thermite and Zr metal to the upper oxide layer of the stratified melt. Current tests in MOCKA (KIT, Germany) program are focused on the erosion behaviour of the LCS concrete. Hence, a series of MOCKA experiments have been conducted with and without rebars [35]. The melt is generated in a thermite reaction and the decay heat is simulated by adding the thermite briquettes and Zr into the melt from the top. In cooperation with CEA, a large-scale test on efficiency of modified COMET core-catcher concept was successfully performed addressing the issue of coolability of ex-vessel melts in the reactor cavity [36].

These test facilities contributed to a better understanding of the core melt sequences and thus improve safety of existing and, in the long-term, of Gen-III and –IV reactors by severe accident mitigation measures and by safety improvements where required. They are part of different European projects such as SAFEST [37] and ALISA [38] projects of the 7th EU Framework Programs.

2.3.2 KIT numerical investigations for severe accidents

The KIT theoretical investigations are focused on improvement and validation of severe accident codes such as ATHLET-CD, ASTEC and MELCOR in the frame of national projects (WASA-BOSS) or international projects (CESAM, FASNET). Recently, severe accidents investigations for both PWR and BWR aiming to improve the SAM guidelines based on the insight gained from the simulations [39] and [40]. These investigations were driven by the Fukushima accident. It became evident that there is a need to further improve the severe accident management guidelines (SAMG) and strategies to control severe accidents [31] and [2]. Within the CESAM project, potentials of SAM-measures to prevent or delay the failure of the safety barriers such as the reactor pressure vessel of a German PWR konvoi were investigated using the ASTEC code and considering different accident scenarios e.g. MBLOCA, SBO and SBLOCA [41] and [42]. Before the codes were applied for SAM-optimizations, selected in-vessel phenomena were validated using KIT-tests such as CORA-16 (BWR) and QUENCH-8 (PWR) [43].

In Figure 9, the comparison of the measured and predicted hydrogen mass generated during the QUENCH-8 test is shown. The new ASTECV2.1 version predicts the integral value close to the measured data due to a better axial nodalisation and a better modelling of the radiation heat transfer in the upper unheated test section zone. In Figure 10, it can be seen that in case of the Station Black-out sequence a high pressure sequences is established after the loss of AC power due to the rapid reduction of the water inventory of the SGs becoming depleted around AC-power lost. Hence the primary circuit lost his function as heat an effective sink. As a consequence the temperature of the water and the pressure in the primary circuit starts to increase up to the set point of the pressurizer safety valves, where it remains until RPV failure.

Further KIT activities on severe accidents are the following: a) KIT is actively participating in the currently running OECD TCOFF project on thermodynamic characterization of fuel debris and fission products based on scenario analysis of severe accident progression at Fukushima Daiichi NPP b) KIT was also involved in the already completed OECD BETMI-2 project on ability of current advanced codes to predict in-vessel core melt progression and degraded core coolability and c) Analysis of the Fukushima Unit 2 accident with the MELCOR code in cooperation with the Polytechnic University of Madrid.
2.4 Emergency management and long term rehabilitation

KIT activities in this field are concentrated on the development of the RODOS (Real-time On-line DecisiOn Support) system for off-site emergency management after nuclear accidents. Its development was supported by the EC in the frame of the Euratom Research Framework Programme for more than two decades. The RODOS system covers now all phases of an emergency and can be applied worldwide. The current version named JRodos is the result of an engineering process initiated by the user community. It is fully JAVA based and can be operated under Windows and Linux as the JAVA structure allows such independent installations [44]. Recently, more than 20 institutions in 16 countries operate JRodos at national and local level e.g. Austria, Finland, the Netherlands, Switzerland, Ukraine, Hong Kong in addition to Germany. It is based on the RODOS system that is a non-commercial real-time on-line decision support system for off-site nuclear emergency management that provides consistent and comprehensive information at local, regional and national levels, during all phases of a real event and while preparing for a possible future event [45]. In case of a real accidental event, the system will house all relevant information on the release and the environmental contamination, and it will forecast health, agricultural, economic impacts with and without the application of countermeasures. It can also assist decision makers in evaluating different measures against a range of quantitative and qualitative criteria. The Fukushima disaster in 2011 revealed gaps in emergency management and long term recovery in different European member states. Hence, a European research project named PREPARE (Innovative integrated tools and platforms for radiological emergency preparedness and post-accident response in Europe) started coordinated by KIT in 2013. It aims to close gaps that have been identified in nuclear and radiological preparedness in Europe following the first evaluation of that disaster. One of the most crucial deficits in the management of such accidents are missing or un-complete source term estimations. In different EU projects such as FASNET, the JRodos system is being extended to be able to have an interface for the radiological source term prediction provided by any severe accident code such as ASTEC, MELCOR or ATHLET-CD [44]. KIT is also leading the development of an “Analytical Platform” that can be used for information gathering, processing and communication to all interested stakeholders in the frame of the NERIS-TP project (Towards a self-sustaining European Technology Platform (NERIS-TP) on Preparedness for Nuclear and Radiological Emergency Response and Recovery).
Last but not least, the JRodos system is very much appropriate for training and exercising the personnel and stakeholders that would be involved in the management of severe accidents since it allows creating accident scenarios and the background material needed. For more information please visit: https://resy5.iket.kit.edu/JRODOS/.

3 KIT education and training programs

Education & Training activities on the field of reactor safety and related topics are given in the frame of the School of Energy, see Figure 11. The graduate school comprises the students and doctoral students. Regular courses for under graduate and graduate students offered by the Institute of Fusion and Reactor Technology (IFRT) together with other institutes e.g. the Institute of Neutron Physics and Reactor Technology (INR) and the Institute of Nuclear and Energy Technologies (IKET) within the Mechanical Engineering Faculty. Students can be specialized in energy technologies including nuclear engineering in the master program. The professional school comprises dedicated education and training programs e.g. the AREVA Nuclear Professional School (http://www.anps.kit.edu/) and the Frederic Joliot Otto Hahn (FJOH) Summer School (www.fjohss.eu).

In addition, the KIT institutes are hosting foreign students in the frame of different programs such as Erasmus, Leonardo da Vinci, DAAD-fellowships, etc. and doctoral students financed by partners e.g. Industry, DAAD, EU projects or national initiatives of different countries.

Figure 11 The Education and Training pillars of the KIT School of Energy

Hereafter, a short description of selected programs will be given.

3.1 Frederic Joliot Otto-Hahn Summer School

The Frédéric Joliot / Otto Hahn Summer School is jointly organized by CEA (France) and KIT. Several KIT senior scientists are engaged in the preparation of the curriculum and are represented in the scientific board, the executive bureau and the scientific secretariat. It was founded in CEA in 1995 to promote knowledge in the field of reactor physics in Cadarache, France. Since 2001 is jointly organized by the Nuclear Energy Division (DEN) of the CEA and Institute of Neutron Physics and Reactor technology (INR) of the Karlsruher Institute of Technology, Germany. The lecturers are carefully selected worldwide from internationally leading universities, industry, research institutions and utilities.
Each year juniors as well as experienced scientists and engineers from R&D laboratories, nuclear industry and utilities from Europe, Asia, Latin America and East Europa participate in the summer school.

The main objectives of the summer school are listed below:

- Address the needs and challenges of nuclear systems including reactor and fuel design and operation, evolutionary and innovative concepts, optimal solutions for the back-end of the fuel cycle, etc.
- Keep scientists and engineers abreast of the latest developments in basic and applied nuclear sciences, so as to preserve the high standard of knowledge in reactor physics and reactor technology
- Foster networking and encourages informal discussions and the exchange of knowledge between lecturers and participants

More detailed information about the yearly programs can be found at: www.fjohss.eu.

3.2 AREVA Nuclear professional School

The main goal of the AREVA Nuclear Professional School of KIT/IKET is to preserve the high level competence on nuclear technology in Germany and provides an excellent complementary education program for the needs of the industry. The combination of sound scientific teaching and practical industrial experience presents optimal prerequisites for cutting-edge research. There are different kinds of courses for qualification; e. g. two-years graduate education programs in combination with specified training courses. All training courses can also be attended individually. Master and doctoral research studies can be also performed at the KIT-institutes on different reactor safety topics. A list of the numerous courses offered by the AREVA Professional School can be found at: http://www.anps.kit.edu/.

3.3 The EU GENTLE Project

The GENTLE coordination and Support Action is a joint effort by leading academic and research institutions in Europe to coordinate an E&T program in the field of nuclear fission technology. It consists of 11 partners and it was coordinated by TU Delft. The consortium has as goal to create a sustainable lifelong E&T program in the field of Nuclear Fission Technology that meets the needs of the European stakeholders from industry, research and technical safety organisations. For more details, see http://gentleproject.eu/.

The GENTLE project aimed at the successful implementation of the following joint E&T tools [46], [47]and [48]:

- Student Research Experiences (SRE, mobility) to facilitate students from the participating universities to get hands-on experience in Europe's unique and specialised laboratories and participate to cutting-edge research and student internships (SIs) in research and industry, increasing the value of the students' curriculum significantly.

- Intersemester (ISC, advanced learning) courses for graduate and post graduate students on special industry related topics, which will be provided by academics and specialists from research and industry. The intersemester courses give students an unique opportunity to deepen specific aspects in the field of nuclear science and technology that are not common in the MSc programme of many universities, such as for example safeguard, decommissioning, nuclear waste, reactor techniques and
thermal hydraulic phenomena. The opportunity to work in hot laboratories and controlled areas with radioactive material outside the universities is very unique. The following ISC were developed: a) Nuclear Fuels (JRC Karlsruhe) b) Nuclear Safeguards and Security (SCK•CEN) c) Nuclear Waste Management (KIT-INE) d) Decommissioning (UMAN) e) Nuclear data (JRC Geel) f) Reactor physics skill at research reactor (BME) g) Thermal hydraulics phenomena (LUT).

- A Massive Online Open Course (MOOC, basic learning) on Nuclear Energy for students at bachelor/master level in the technical sciences. The goal was to enhance the knowledge of nuclear reactors and fuel cycles of young professionals working in, among others, industry, consultancy companies or regulatory bodies. The MOOC will provide a general introduction to nuclear energy, both from a societal, economical and technical point of view. Each theme consisted of a number of video-lectures by GENTLE teachers, on-line exercises, and a discussion forum. The MOOC was completed by an online test and the possibility to get a proof of attendance (verified certificate). It was MOOC was offered on the edX platform through the Delft Extension School (https://online-learning.tudelft.nl/courses/understanding-nuclear-energy/). More than 4000 learners registered for the course, about 1000 learners started, and 400 stayed active during the complete course. 65 learners paid and received a verified certificate for the course attendance, of which 50 completed and passed the MOOC. The following online courses were developed: a) Fundamentals of nuclear science (JRC Geel) b) Nuclear fission reactor principles (TU Delft) c) Light water reactor systems and safety (KIT) d) The nuclear fuel cycle (JRC Karlsruhe) e) Life cycle analysis and social aspect/impact of nuclear energy (SCKCEN) f) Next generation nuclear power (CIR TEN).

KIT involvement was as follows a) Work package leader of Intersemester courses and b) Responsible for Module 3 of MOOC devote to “reactor systems and safety”. In 2016, KIT hosted and supervised in total six GENTLE PhD students or young researchers who stayed at KIT for a period between one and six months each.

3.4 The EU ANNETTE Project

The EU project is ANNETTE (Advanced Networking for Nuclear Education and Training and Transfer of Expertise) aimed at promoting the coordinated cooperation of all the nuclear stakeholders in the planning and implementation of education and training and thus to assure qualified workforce in the next decades within Europe. It consists of 25 partners among others KIT. The main aim of this action is to consolidate and better exploit the achievements already reached in the past and to tackle the present challenges in preparing the European workforce in the different nuclear areas, with special attention to continuous professional development, life-long learning and cross border mobility. It is organized in eight work packages such as a) Survey and coordination of networking in nuclear E&T and VET in nuclear areas b) design and implementation of coordinated E&T an VET efforts e.g. masters and summer schools for continuous professional development c) Generational transfer of expertise d) Cross-border transfer of expertise e) ETI actions related to nuclear safety culture f) Coordination of the nuclearization of fusion g) link with stakeholders and h) Management. For more details, see www.enen-assoc.org.
4 Summary

This paper summarizes the KIT infrastructure of both experimental facilities and numerical simulation tools for reactor safety and technology built and developed during the last two decades. It covers both the design basis and severe accident research areas including emergency management platforms. In addition, the KIT education and training activities concentrated within the School of Energy are presented. Selected experimental facilities were described and also results of selected simulations are shown to demonstrate the unique combination of experimental investigations and numerical simulations in the fields of reactor safety. Generally it can be stated that KIT is operating unique test facilities and developing, validating and applying simulation tools which are very much appropriate for any educational and training purposes to increase the competence in safety-related areas of expertise needed to assess the overall safety of nuclear power plants.

References


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