ANMT 2019

Key Topic: Outstanding Know-How & Sustainable Innovations

Technical Session: *Development, Validation and Application of the GRS Code System AC² for Currently Operated and Future NPP*

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The technical session *Development, Validation and Application of the GRS Code System AC² for Currently Operated and Future NPP* was chaired by *Andreas Schaffrath* and *Sebastian Buchholz* (both from Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH). It was well attended with up to 30 listeners, who actively contributed to the session with questions, remarks and comments.

As *Andreas Schaffrath* explained in his short introductory statement, over 60 technical experts of the safety research division of GRS are developing and validating reliable methods and computer codes – summarized under the term *nuclear simulation chain* – for the safety-related assessment for all types of nuclear power plants (NPP) and other nuclear facilities considering the current state of art science and technology. The *nuclear simulation chain* shall be able to simulate and assess all relevant physical processes and phenomena for normal operation, anticipated operational occurrences, design basis accidents, and severe accidents in nuclear power plants (NPPs). Among other applications the *nuclear simulation chain* is used in nuclear licensing and supervisory procedures for independent assessment of evidences submitted by NPP operators. Numerous national as well international organisations (e.g. universities and research centres) support GRS in the development and validation. The GRS codes can be passed on request to other (national as well as international) organizations. This contributes to a worldwide increase of the nuclear safety standards. The code transfer is especially important for developing and emerging countries lacking the financial means and/or the necessary know-how for this purpose.

The technical session focused on the central element of the GRS simulation chain: the code System AC², which consists of the thermal-hydraulic code ATHLET, the severe accident code ATHLET-CD and the containment code system COCOSYS. AC² also includes effective and powerful software tools for pre- and postprocessing (e.g.

ATLASneo). The session contained eight outstanding presentations from different nuclear stakeholders and underlined impressively the large application spectrum as well as importance of AC² for nuclear safety research in general and currently operated and future NPP particularly. Thus, AC^2 is a central component for the preservation as well as expansion of the nuclear competences in Germany.

As first speaker, *Fabian Weyermann* (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH) presented the **Development Strategy of the GRS Code System AC²**. The focus of his presentation was the motivation for the AC² development, the new AC² release and future developments and improvements. The motivation for coupling of the codes ATHLET, ATHLET-CD and COCOSYS to the code system AC² are the new simulation requirements of the advanced (Gen III/III+) reactors, the innovative (Gen IV) reactors designs and the small modular reactors (SMRs). These are e.g. the passive safety systems, new innovative components (e.g. bayonet, plate or helical coil steam generators), new working fluids (e.g. supercritical water, liquid metals, gases, molten salts), new containment concepts with an infinite passive containment cooling to an ultimate heat sink which could be either air or water and finally large water pools. The new release announced for July 2019 will include numerous new features. These are besides extensive model improvements an improved coupling interface, a new numerical tool kit (NUT), a new ATHLET Input Graphics (AIG) and GCSM modeler (AGM). Finally, the goals for future code developments were presented. Here the focus lies on the improvement of the code performance and the pre- and postprocessing, the integration of an efficient uncertainty analysis and the completion of the models for the advanced reactors, the innovative reactor designs and the SMRs. The latter requires specifically the simulation of natural circulation flows with small driving pressure differences often at low or near vacuum pressures. Further development priorities are coupled multi-physics calculations (e.g. considering 3D fluid dynamics or neutronics and/or structural mechanics).

Next, *Sebastian Buchholz* (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH) presented selected aspects of the **Validation of the AC² Modules ATHLET and ATHLET-CD.** He started with an overview on validation activities and presented afterwards selected results of the validation of ATHLET and ATHLET-CD. The basic procedure of the development and validation of the code of the GRS simulation chain is described in a QM guideline, which is based on international practice. In general, development and verification on the one hand and validation on the other shall be carried out by different persons. Verification denotes the control, whether correlations and models are implemented correctly in the code, while validation describes the review process ensuring that the physical phenomena are correctly described by the models. Each validation starts with the preparation of suitable text matrices, which includes a reactor design specific compilation of relevant phenomena and test facilities, in which these were investigated. The test matrices have been jointly developed by different code developers and published by the Committee on Safety of Nuclear Installations (CSNI) of the Nuclear Energy Agency of the Organization for Economic Co-Operation and Development (OECD/NEA). The test matrixes contain single effect tests for the investigation of selected phenomena under clearly defined and adjusted initial and boundary conditions with a high instrumentation density and quality as well as integral tests. Here interactions of different phenomena are investigated in scaled test facilities. After a sound overview on current validation activities, selected validation results (ROCOM3.1, PHEBUS FPT3, PKL Test I2.2, QUENCH19) were presented and discussed. *Sebastian Buchholz* concluded that the results of the afore mentioned validation calculation show that the modules ATHLET and ATHLET-CD can be successfully applied forthermal-hydraulics of LWR Gen II, III and IV, core degradation, fission product release and transport as well as late phase phenomena.

In the third presentation *Kai Kosowski* (Preussen Elektra) reported on the **Application of a NPP Analysis Simulator based on ATHLET – Robustness Analysis of an External Hazard coinciding with SB-LOCA.** He stated that the safety behavior of an NPP after an external hazard-initiated event, as well as after a small break LOCA are already well-known as part of the analyses done for standard license application. The coincidence of both events leads to a beyond-design basis accident (BDA), which was investigated by means of the thermal-hydraulic system code ATHLET within ATLAS simulation environment, both developed by the Gesellschaft für Anlagen- und Reaktor-sicherheit (GRS) gGmbH. The scenario assumes an external event with a LOCA caused by induced vibrations on a small pipe attached to the primary circuit, although all pipes are designed to withstand such loads. Furthermore, in the context of both robustness and enveloping analyses, a loss of offsite power (LOOP) and the unavailability of the emergency diesel power supply due to the external impact are postulated. For emergency cooling, the NPP in the scenario considered has only access to the passive accumulators and to systems supplied via the safeguard emergency diesel engines (2nd quartet of emergency diesel engines) which are housed in the bunkered emergency feed building. The scenario was modelled and simulated with an NPP analysis simulator based on ATHLET. The results show that the remaining systems for emergency cooling were able to handle the LOCA under such demanding conditions. Most importantly, the heat removal from the core was sufficient during the entire time. Eventually, it could be demonstrated that all safety protection objectives and targets were met for this beyond-design basis scenario.

The next two lectures were given by *Rafael Macian-Juan* (Chair of Nuclear Energy of the Technical University of Munich). The first one has the title **Study of the Suitability of ATHLET for the Simulation of a Molten Salt Reactor (MSR).** The Background for this work was, that MSRs, one of the Generation-IV reactor concepts, are becoming in countries such as USA, EU, Russia and China an increasingly interesting option because of their robust operating characteristics and fuel handling capabilities. The objective of this work was to show that ATHLET could be used to simulate the thermal-hydraulic and neutronic behavior of a MSR reactor after appropriate modifications were implemented. *Rafael Macian-Juan* started with a short introduction of MSRs. After that he summarized relevant ATHLET features for MSR simulation. He summarized that ATHLET is e.g. capable to

- accurately simulate complex MSRs flow configurations,
- calculate the heat transfer between solid surfaces and the fluid(s),
- model the neutronic behavior with temperature and density feed-backs,
- simulate special operating procedures in MSRs,
- model the reactor, plant and fuel processing control systems.

Rafael Macian-Juan continued that TUM has implemented new physical properties for the molten salt in the Molten Salt Reactor Experiment (MSRE) built at Oak Ridge National Laboratory (ORNL) constructed by 1964, went critical in 1965 and operated until 1968. Additionally, the point-kinetics equations and their integration were modified. Afterwards he presented re-calculations of steady state and transient tests, which show a fairly well agreement with the experiments. The deviations are in the same order of magnitude as the deviations of the TRACE calculations.

The second contribution of *Rafael Macian-Juan* (Chair of Nuclear Energy of the Technical University of Munich) has the title **Uncertainty and Sensitivity Analysis of the Molten Lead Experiment TALL with the coupled code ATHLET-ANSYS/CFX.** The author stated that the TALL-3D facility, built by KTH in the scope of the THINS project, aims at investigating challenging phenomena in a facility filled with lead–bismuth eutectic (LBE) containing a pool. *Rafael Macian-Juan* explained that the experimental data were used for the validation of the TALL-3D facility models developed by the partners of the SESAME project. Based on the coupling between ANSYS CFX (CFD) and ATHLET (system code) implemented by the GRS, TUM performed an uncertainty and sensitivity analysis of the benchmark test TG03.S301.03. In this analysis the uncertainty in the output which is due to the uncertainty on the input (uncertainty analysis) was investigated and the influence of the uncertain parameters (sensitivity analysis) was assessed. The benchmark was performed in two phases. For the blind phase, in which the participants do not know the experimental results, large deviations between experimental and computational data were observed. The predictions with the tolerance limits could not envelop the experimental data. This led to the conclusion, that an improvement of the simulation model was needed. In the second open phase of the benchmark the experimenters from KTH provided the measured data to the participants. After further calibration of the simulation models (e.g. pressures loss coefficient in the pump and heat losses in primary circuit) the simulation results were significantly improved. From the benchmark exercise two major conclusions could be drawn: first uncertainty and sensitivity analyses are essential for identifying influential and uncertain input parameters in the computer models. Second, the blind benchmark exercise shows the currently lacking maturity to predict safety related thermal-hydraulic phenomena in nuclear installations, for which no (operational) experience is available.

Next, *Aurelian Badea* (of the division Innovative Reactor Systems of Institute for Applied Thermal-Fluidics) presented the **Application of the ATHLET Code to Super-critical Water-Cooled Systems**. The author started with on overview on the KIT database containing more than 53000 experimental data for the heat transfer of supercritical water in tubes, annuli and rod-bundles. Then *Aurelian Badea* introduced a methodology to assess the intrinsic consistency of the 28364 experimental data points for supercritical water heat transfer in the tubes. The application of this methodology showed a good overall consistency. Afterwards new heat transfer correlations were developed for large domains (flow parameters, geometries) and based on 2 different types of boundary conditions (given heat flux or given wall temperature). He stated that these correlations describe very well the supercritical heat transfer for the tubes (85% of the respective data are predicted with an accuracy better than 20%). These correlations can be also applied for the supercritical heat transfer in annuli (90% of the respective data are predicted with an accuracy better than 30%) and rod bundles with spacer grids (85% of the respective data are predicted with an accuracy better than 40%). The new correlations were implemented in ATHLET which is now able to precisely calculate the heat transfer in the region of the pseudo-critical temperature, especially with given surface wall temperature. After modification the Westinghouse W-3 correlation in the pressure range from 17 – 21 MPa also the boiling crises could be simulated.

In the seventh lecture *Michael Buck* (Institute of Nuclear technology and Energy Systems, Stuttgart) presented **ATHLET Extensions for the Simulation of Supercritical Carbon Dioxide Driven Power Cycles**. The supercritical carbon dioxide (sCO_2) driven power cycle is a new and very compact approach for an active heat removal with a turbo compressor. It uses air as an ultimate heat sink and is independent from access to electrical grids. Afterwards the author described the implementation of such a system in a BWR, first orienting analyses and a demonstration installation in the glass model of the Gesellschaft für Simulatorschulung (GfS) in Essen. Then necessary tasks for the simulation of sCO_2-Bryton Cycles and its interactions for advances LWR are summarized. These are code extensions (such as the implementation of improved thermo-dynamic property data for sCO_2 and models for radial turbomachinery and compact heat exchangers), code validation (e.g. the compact heat exchangers against experiment performed at IKE and the glass model of GfS). Finally, *Michael Buck* gave an outlook for future work. For his point of view the implemented equations of state for sCO_2 need further extensions (e.g. for liquid, vapor and metastable region). Further extensions are needed for turbomachinery (considering of experimental data recorded in the meantime) and the heat exchanger (cross flow and cooling) modelling. The correlations for heat transfer and pressure losses of CO2 – especially close to the critical point – require further validation. Finally, systems calculations especially for transient behavior and interactions with other systems shall be performed.

The last presentation given by *Tobias Jankowski* (University of Bochum, working group Plant Simulation and Safety) deals with **Challenges in Modelling and Validation of Wet Resuspension Phenomena**. He stated that wet resuspension phenomena are of interest in the (late phase) source term estimation. This is especially valid for postulated severe accidents with code degradation and RPV failure, in which e.g. the melt in the spreading compartment is flooded with sump water. The sump water may contain washed out/down fission products. Then *Tobias Jankowski* pointed out, that there are many uncertainties in wet resuspension data acquisition. Therefore, a lot of basic research focuses on measurements of single bubble ruptures. The respective data can be reproduced by OpenFOAM calculations. Afterwards he presented flow regime dependent modeling concepts for the estimation of the superficial gas velocity for subcooled and boiling water pools and for droplet entrainment. For boiling pools the wet resuspension correlation already presented at the 49th AMNT, 2018 in Berlin was implemented in COCOSYS. Its application leads to a significant improvement of the simulation results. For subcooled pools a stepwise validation is still ongoing.

At the end of the session the organizer *Andreas Schaffrath* thanked all the speakers for their vivid, interesting and impressive presentations. They illustrated the enormous range of application of the GRS Code System AC². AC² benefits immensely from the external developments and validation. This external work is therefore essential to consolidate AC² as one of the world’s leading thermal-hydraulic code system e.g. as evidence tool for nuclear procedures for regulators and TSOs as well as platform for nuclear safety research e.g. performed by universities and research centres/organizations. Therefore, AC² is a central element for maintaining and further building-up nuclear competences in Germany.