Analyses in Regulatory Practice

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Abstract:
The application of numerical codes is well established in the regulatory practice for nuclear power plants. Therefore, requirements for computer-aided analyses can be found in different places of the nuclear technical rules and standards. These requirements also apply for CFD analyses which reached more and more the spot light of research activities and component design, because computing capacities increased significantly in the past years. Hence, the actual status of CFD as a possible design and analysis tool in the supervising process of nuclear power plants will be discussed in this paper. The focus is set on the prerequisites which have to be fulfilled by codes and in particular by the user who wants to use CFD methods. The prerequisites will be derived directly from the existing German technical rules and standards. Finally, two examples for typical CFD applications for nuclear safety analyses are given.

1 INTRODUCTION

The use of numerical software is well established in many applications in nuclear technology. For instance, there is a long-time experience with codes in the fields of neutron kinetics, structural mechanics or thermalhydraulics. The codes are not only applied to dimension and to design systems and components of nuclear power plants. Moreover, they are used for safety analyses of transients and incidents. Therefore, the numerical methods and tools have to be very reliable and they must fulfill strict requirements to prove their suitability. But facing that many misleading numerical results are caused by wrong usage also the users have to deal with a large responsibilty.

Computational Fluid Dynamic (CFD) codes were and still are developed for the simulation of multidimensional flows as they occur in the reactor cooling system or the containment of nuclear power plants. Their advantage is their ability to resolve local flow phenomena also in complex geometries. Thanks to increasing computational resources they nowadays offer the possibilty to simulate complex flows more realistically than other numerical methods. In fact a comprehensive database of knowledge and experience already exists for selected single phase flow phenomena. But in particular multiphase and multicomponent flows are not understood completely or still beyond the computational means. Thus, there is an ongoing demand and research for numerical methods and validation.

A modern strategy is the multi-scale approach which means a combination of CFD simulations for different length scales. The idea is to gain more insights in the local physics of complex flows, e.g. multiphase flows, by numerical simulations in the micro-scales without any simplifying assumptions, so-called direct numerical simulations. Due to the turbulent character of industrial flows and the resulting computational effort this is only possible for very small domains and generic problems, e.g. coalescence of only a few steam bubbles. Based on the findings in micro-scale new CFD models are going to be developed for larger scales which make use of simplifications like averaging, turbulence or multiphase models. At the end of this chain are system-scale codes like ATHLET which simulate entire systems and plants and abstract from local flow phenomena. A promising approach is to couple these system codes with CFD methods to compute local flow phenomena only in those parts of a system where it is necessary. Altogether there exist several different methods which all depend on reliable and adequate CFD analyses.

What makes the situation even more complicated is the fact that an insufficient user input may lead to erroneous results as it was already mentioned before. Therefore, a reliable usage of CFD methods and a specification of uncertainties can be assured only through the application of so-called best practice guidelines which is elaborate and sometimes not possible for practical reasons.

Against this background it will be discussed in the following what is necessary for using CFD methods for analyses in the nuclear supervising procedure. In particular, the requirements for numerical analyses will be worked out from the technical rules and standards. Finally, two examples for typical CFD applications are described at the end of this paper, namely boron dilution transients and pressurized thermal shocks.

2 REQUIREMENTS FOR THE APPLICATION OF NUMERICAL METHODS IN REGULATORY PRACTICE

As part of the nuclear supervising process the state authorities may consult technical expert organisations like TÜV NORD SysTec to assist them in the review process of reports submitted by the operators of the nuclear facilities [3]. The experts check whether the analyses fulfill all safety requirements with regard to the state of science and technology. For this purpose the experts frequently use computer-aided analyses as the operators do for their reports. Commonly-used numerical codes are, e.g. CASMO/SIMULATE [13], [14] to perform neutron kinetic analyses and ATHLET [6] or S-RELAP [12] which are used for system-scale thermalhydraulic analyses. Furthermore, INROS and DYVRO [15] are codes which are successfully applied for pressure surge analyses and ANSYS [2] is well-known as a tool for structure mechanical problems. All codes which are used in the regulatory practice have in common that they are qualified and validated for their individual purposes. Generally, they have proven their suitability in the field over many years.

The aforementioned requirements for a reliable application of numerical software concern also the user. Particularly, the verification that the CFD analyses are based on a reliable fundament demands a great effort from the user because of the complexity of the simulated phenomena. This is expressed by many recommendations and best practise guidelines [9].

Essential principles for a reliable application of numerical software are a successful

- verification and
- validation of the code as well as
- the specification of remaining uncertainties.

Verification means the confirmation of the correct functionality of the software in accordance with its specification. A code can be verified for instance by comparing its results for simple test cases with known solutions (e.g. analytical solutions). In general, this has already been done by the author or distributor of the software. However, the user has to be able to confirm the verification. Therefore, an appropriate documentation of the test cases and the implemented algorithms is indispensable.

The *validation* answers the question whether a method or model is suitable for a specific application. This is important because the simulation of complex physical processes - like e.g. turbulent flows - requires in general simplifying assumptions. Therefore, the simulation results have to be compared with experimental data from representative measurements. In this context it has to be distinguished between recalculation and forecast of an experiment. Especially since the first case is often used to adjust some modelling parameters. At the end the method/model should be able to reproduce the experimental data with sufficient accuracy and without any further adjustments. However, it has to be considered that already the experimental data usually have some measurement uncertainties.

Uncertainties affect not only measurements. In fact, all numerical methods contain different sources of uncertainties like, e.g. modelling uncertainties, iteration as well as discretization errors and so on. Therefore, the identification and evaluation of uncertainties is crucial for the assessment of the reliability of numerical results.

The requirements for numerical analyses which apply analogously also to CFD simulations can be found at different places in the technical rules for nuclear applications, e.g.

- national:
  - guidelines and recommendations of the Reactor Safety Commission (RSK),
  - guidelines of the Nuclear Safety Standards Commision (KTA),
  - the Safety Requirements For Nuclear Power Plants, Annex 5 “Requirements for Safety Demonstration and Documentation”
- international:
  - the IAEA Safety Report “Accident Analysis for Nuclear Power Plants” [7] and
  - US NRC Regulatory Guide 1.157 “Best Estimate Calculations of Emergency Core Cooling System Performance” [16].

The following passages go into the details of some selected extracts from these rules and standards.

The *guidelines of the Reactor Safety Commission* for Pressurized Water Reactors [10] contain in chapter 22.1.3 “Assumptions for Emergency Core Cooling Calculations” instructions for emergency core cooling analyses. First of all the importance of validation is stressed by the requirement that experimentally verified analyses have to be submitted which confirm the effectiveness of the emergency core cooling for all relevant operation conditions. Furthermore, chapter 22.1.3 contains in the paragraphs 1 to 14 detailed specifications for conservative boundary conditions. Paragraph 1 for instance gives specifications for discharge flow rates, paragraphs 3, 4 and 6 consider the heat flux between cladding tubes and coolant, paragraph 8 contains instructions for pump behavior and paragraph 12 deals with the power distribution in the core.

The *recommendation of the Reactor Safety Commission* “Anforderungen an die Nachweisführung bei Kühlmittelverluststörfällen” [11] deals also with the requirements for computer-aided analyses. It considers in particular the uncertainties of such analyses. The Reactor Safety Commission distinguishes two cases, so-called “best-estimate” analyses and a simplified approach. The best-estimate approach applies models which are as realistic as possible and which have to be successfully validated to prove their suitability. Since the available experimental data is frequently based on scaled experiments, the Reactor Safety Commission points out that the validation has to consider also scaling effects. Although best-estimate analyses aim at reproducing reality in the best possible way, the Reactor Safety Commission recommends some deterministic boundary conditions for this kind of analyses. Much like the instructions in the aforementioned guidelines for Pressurized Water Reactors the intention is to guarantee conservative results. Nevertheless the Reactor Safety Commission considers an uncertainty analysis of models and measurements as essential in the case of best-estimate analyses. Otherwise it is not possible to judge the reliablity of the results. During the uncertainty analysis the uncertainties of the models should be determined by appropriate experiments, whereas uncertainties in the plant status should be considered by probabilistic models. Thus, relevant plant parameters have to be allocated with probability distributions to study their influence. In contrast to best-estimate analyses the simplified approach aims at conservative results rather than realistic ones. Therefore, it is necessary to identify conservative values for sensitive influence parameters. An uncertainty analysis can be omitted in that case.

A more exhaustive overview concerning the application of numerical methods can be found in Appendix B of the *rule 3201.2 of the Nuclear Safety Standards Commision “Components of the Reactor Coolant Pressure Boundary of Light Water Reactors; Part 2: Design and*

Analysis" [8]. The guideline contains instructions for mesh generation or the temporal resolution of a transient analysis. Furthermore, it demands a physical (fulfillment of the conservation equations, plausibilty of the results, …) as well as a numerical control of the results. Usually the partial differential equations are discretized and continuous solutions are approximated at grid points. Thus, the numerical solutions contain discretization and iteration errors. Moreover, the finite precision of computers induce additional rounding errors. Hence, the user has to handle all these possible error sources. Beside these purely technical instructions the guideline also contains requirements for documentation and reliability of the codes. The documentation has to cover the theoretical background of the applied methods as well as verified and comprehensible application examples for the verification of the code. To prove the reliabilty of the software

- a modular code structure,
- a standardized programming language,
- centralized support,
- a large user community and
- frequent application

are demanded by the guideline. Regarding code validation the guideline admits beside a comparison with appropriate experiments also code-to-code comparisons with other validated codes.

On the 22th of November 2013 the federal and state authorities passed the *"Safety Requirements for Nuclear Power Plants"* as a new technical rule for nuclear reactor safety [4]. Annex 5 "Requirements for Safety Demonstration and Documentation" which concerns the realization of saftey analyses summarizes most of the requirements already described in the preceding paragraphs. Although it contains no completely new requirements many topics are explained in more details. For instance it is precisely defined when a code can be regarded as validated and that it is not enough to compare numerical results with experimental data, analytical solutions or other validated codes. Also the range of application has to be checked. If the experimental data doesn't cover the range of application the portability of the data onto the actual range of application has to be shown. Furthermore, requirements for the quality of data and the documentation of the validation are listed. Regarding best-estimate analyses the new rule demands that uncertainty analyses should be performed with a 95 % confidence level and that the acceptance criterion is fulfilled with a probability of 95 %. If it isn't possible to capture uncertainties via parameter variations they should be considered by a surcharge which was derived from the validation. In case of simplified analyses for which uncertainty analyses can be omitted several ways are explained in annex 5 in order to obtain conservative results. Finally, the new technical rule specifies in detail the demanded boundary and initial conditions for specific safety analyses and the respective safety level, e.g. Loss of Coolant Accidents. However, since this is not a complete list of the requirements which are addressed the interested reader is referred to the rule for more details.

Furthermore, the requirement of code validation occurs also in the standard DIN EN ISO 9001:2008 in section 7.6 [5]. Thus, it is not only a demand for nuclear applications. Also the DIN EN ISO 9001:2008 standard requires that the suitability of the software for a specific task has to be proven before its first application.

A concrete guidance for the fulfillment of the requirements for CFD applications is contained in the "Best Practice Guidelines for the Use of CFD in Nuclear Reactor Safety Application" (BPG) [9]. This guideline picks up many of the topics mentioned above and gives assistance for CFD analyses. For instance, it points out possibilities to minimize numerical errors and to avoid modelling errors, respectively.

3 CFD ANALYSES IN REGULATORY PRACTICE

For topics in nuclear safety which require an exact knowledge of local flow phenomena like, e.g. temperature or concentration distributions in complex geometries, CFD analyses are already an alternative to expensive experiments. In particular, if it necessary to perform experiments in original scale to avoid scaling effects. Typical applications are boron dilution transients or the protection of the pressure vessel against brittle failure. Both examples will be discussed in chapters 3.1 and 3.2, respectively.

CFD analyses can replace experiments as a design basis only if the requirements discussed in chapter 2 are fulfilled. However, the ongoing research activities show that there is still a large need for experimental data and new CFD models. Therefore, the usage of CFD analyses is still limited to selected, mainly single-phase applications. "Blind" Calculations without an appropriate validation for every specific application are still exceptions. Using best practice guidelines helps to minimize and avoid errors, but it increases the necessary computational effort. Therefore, experiments or simplifying conservative approaches are used in the daily practice of the supervising process rather than CFD analyses. CFD analyses are usually applied to support the proof of compliance, to clearify open questions or to lead to a better understanding of the underlying physics.

3.1 Boron dilution transients

The first example treats boron dilution transients caused by reflux condenser mode. In Pressurized Water Reactors the decay heat can be transferred from the core to the secondary side in reflux condenser mode during a Small Break Loss of Coolant Accident (SB-LOCA) if the water level in the primary circuit is so low that natural convection is not possible any more. In reflux condenser mode steam from the core condenses in the steam generators and flows in counter-current direction back to the reactor pressure vessel. A portion of the condensed steam passes the u-tubes and reaches the cold side of the steam generators. There it may accumulate in the pump seals of loops without the emergency coolant injection. When the water level rises again and natural convection restarts, the condensate, which contains hardly any boron, is transported to the core. The core needs a minimum amount of boron to stay subcritical. Since the condensate is mixed on its way to the core with boron containing coolant the boron concentration increases continuously, but the question is: What is the minimum possible boron concentration which reaches the core?

Many experiments were conducted concerning this question at different test facilities, e.g. UPTF, PKL and ROCOM. In the following the focus is set on the ROCOM experiments. ROCOM is a 1:5 scaled model of a PWR primary circuit and it uses wire mesh sensors to visualize concentration profiles in the flow. The experiment which is presented subsequently is based on PKL III E3.2. The scenario is a SB-LOCA with emergency coolant injection in only two of four loops. The scenario was simulated at the PKL test facility. PKL is scaled 1:145, but it preserves the original heights. Therefore, it is particularly suitable for experiments with natural convection. In the ROCOM test facility it was assumed that the condensate plugs contain a homogenous boron concentration of 50 ppm. The rest of the coolant contains 2500 ppm. The distance between plugs and pressure vessel corresponds to the distance of the pump seals from the pressure vessel. Furthermore, the volume of the pump seals was used as plug size (cf. figure 1). The time dependent mass flows in the cold legs were derived from the PKL results.

This experiment was simulated with ANSYS CFX 12 [1]. The model includes the four cold legs, the downcomer of the reactor pressure vessel and the lower plenum (see figure 2). The core inlet has been simplified by 193 tubes - one for each fuel assembly. The mesh consists of 3.5 Mio. hexaeder and tetraeder cells and is shown in figure 3.

The black curve in figure 4 shows the time devolution of the minimum boron concentration measured in the experiment. The dashed black curve shows the boron concentration averaged over the core entry plane. The corresponding CFX results are represented by the red curves. A comparison of the curves reveals a stronger fluctuation of the numerical results but the trends of the curves are very similar. However, it has to be noted that the experimental results are ensemble averages over 10 repetitions of the same experiment. To consider this fact also time-averaged numerical results are added in figure 4 (dash-dotted, blue curve). In particular the averaged minimum value of the boron concentration agrees quite well with the experiment.

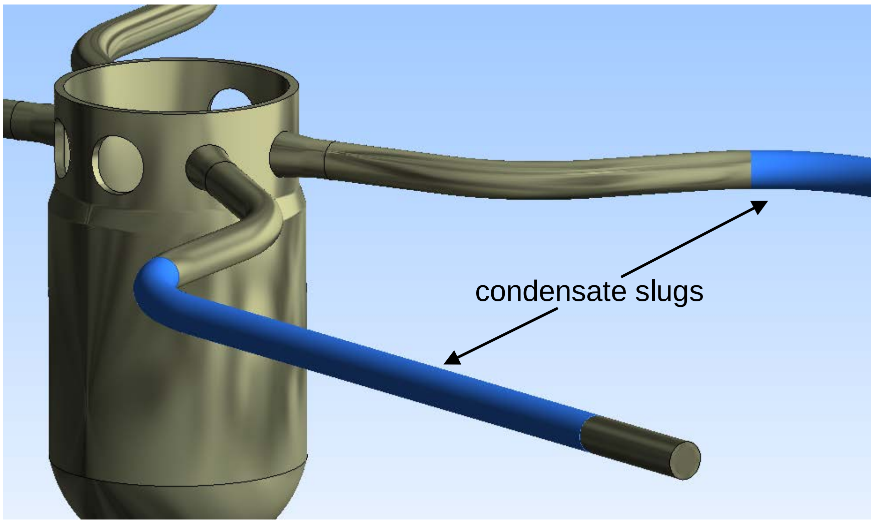


Fig. 1: CFD model of the ROCOM test facility

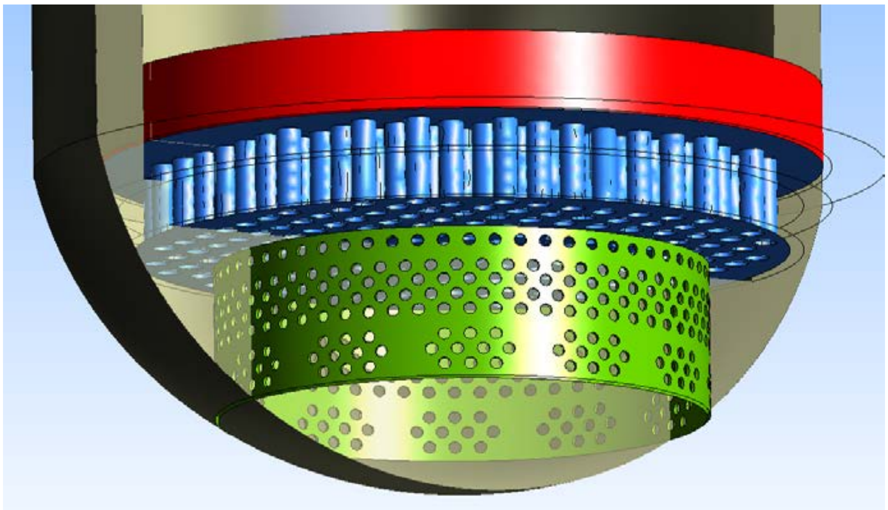


Fig. 2: Lower plenum with perforated drum

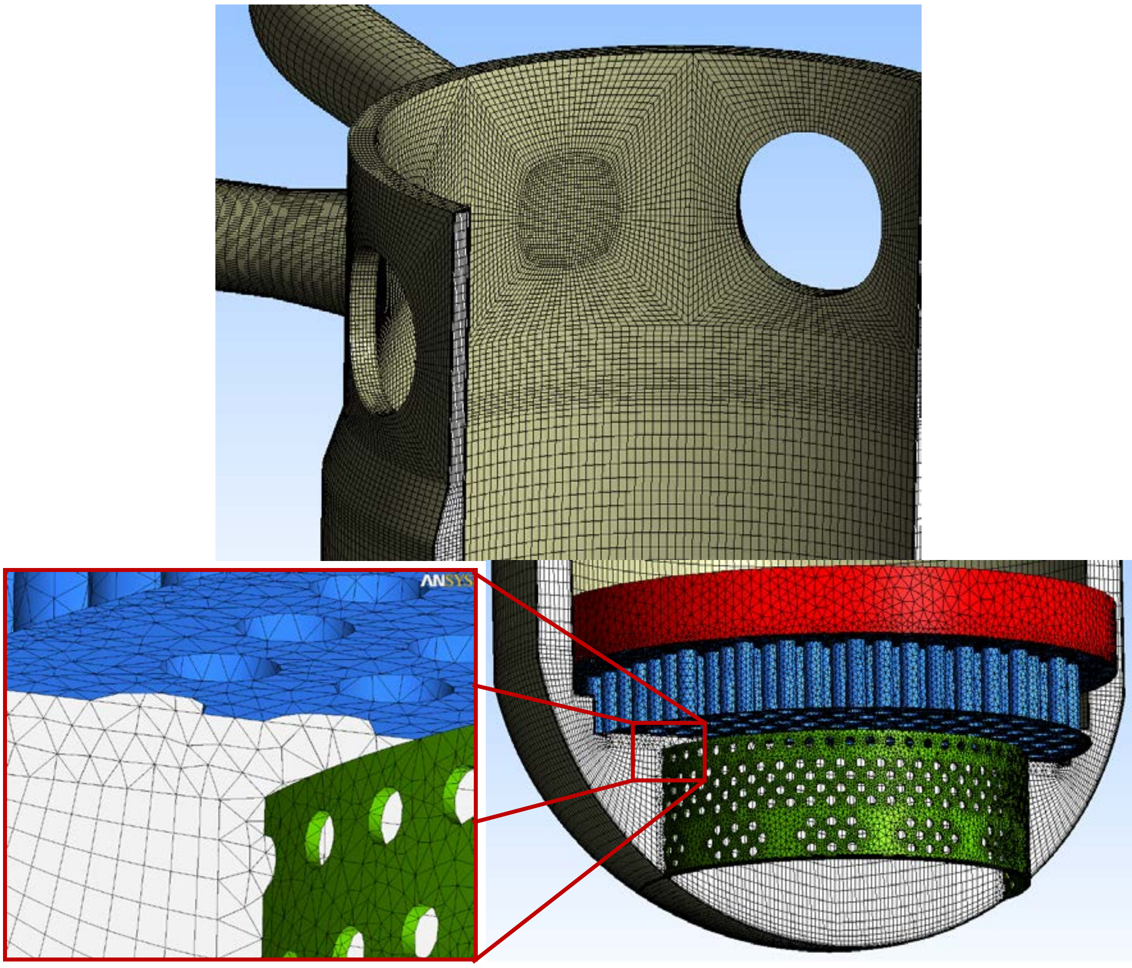


Fig. 3: Snapshots of the hybrid mesh

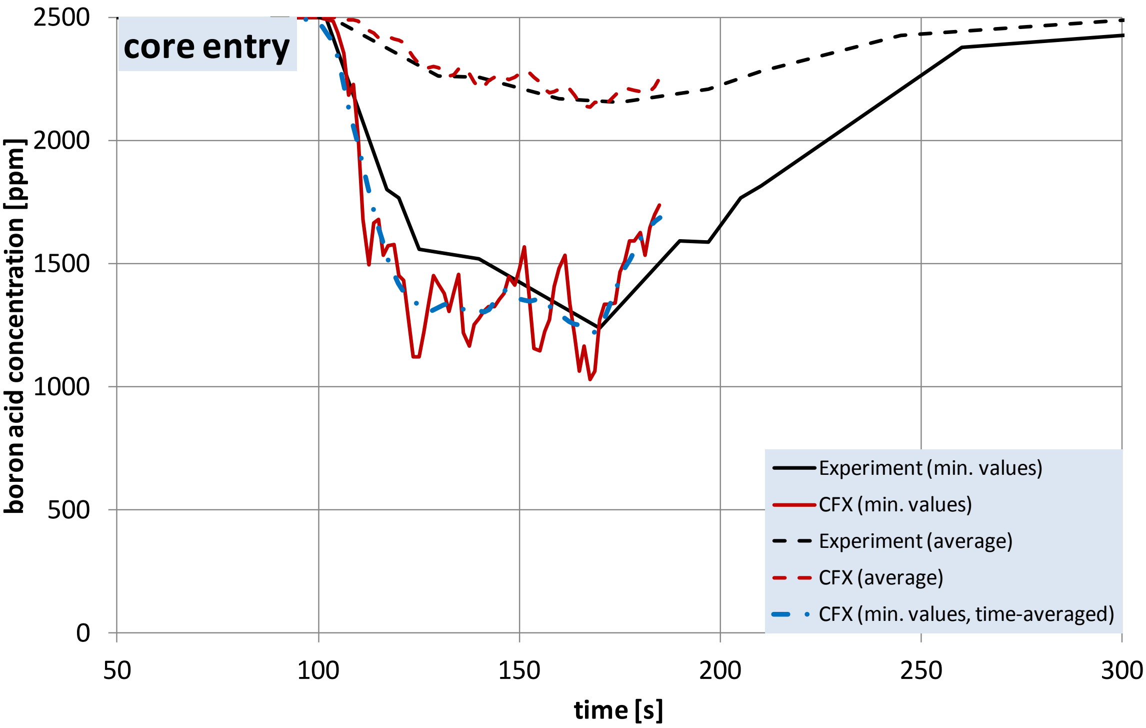


Fig. 4: Boron concentration at the core entry (ROCOM experiment PKL III E2.3)

3.2 Protection of the reactor pressure vessel against brittle failure

The second example considers the protection of the reactor pressure vessel (RPV) against brittle failure. Brittle failure is usually caused by thermal strains and high pressure and characterized by fast growth of small cracks which may end with the complete failure of the vessel.

Causes for brittle failure are high temporal or spatial temperature variations. The fastest temperature variations occur during a guillotine break of a main pipe. But during this transient the coolant is well mixed due to the high injection rates. Thus, this case can be treated efficiently by system codes like ATHLET.

In case of middle or small size breaks cold water streaks arise below the nozzles which belong to legs with emergency coolant injection. Therefore, the attention is drawn to large azimuthal temperature differences which may be determined in particular with CFD simulations.

Figure 5 shows two cold water streaks in the downcomer computed with CFX. The underlying transient is a small break LOCA with emergency coolant injection into two adjacent cold legs.

In these cold legs a stratified flow regime develops. The colder emergency coolant flows at the bottom of the pipe towards the reactor pressure vessel. Above the cold water is warmer coolant. Mixing and warming of the emergency coolant takes place mainly in the vicinity of the injection nozzles. If the break is in addtion in one of the cold legs with coolant injection the injected coolant gets nearly completely lost through the break (see streamlines in figure 6). Figure 7 shows the temperature distribution at the bottom of the cold leg.

Between biological shield and reactor pressure vessel guard pipes surround the main pipes. If cold water discharges through a break near the nozzle it may be led through a small annular shaped gap between guard pipes and nozzle. In this way the coolant reaches the outside of the reactor pressure vessel and flows downwards as a cold water strip which leads to temperature gradients across the wall. This situation is displayed in figure 8. It shows the water volume fraction. A volume fraction of one means pure water, whereas zero stands for pure air.

For this example the actual documents which were submitted in the regulatory process were not based on CFD analyses, but on experimentally verified, analytical models. The preceding CFD analyses served only during the assessment of TÜV NORD mainly for phenomenological investigations.

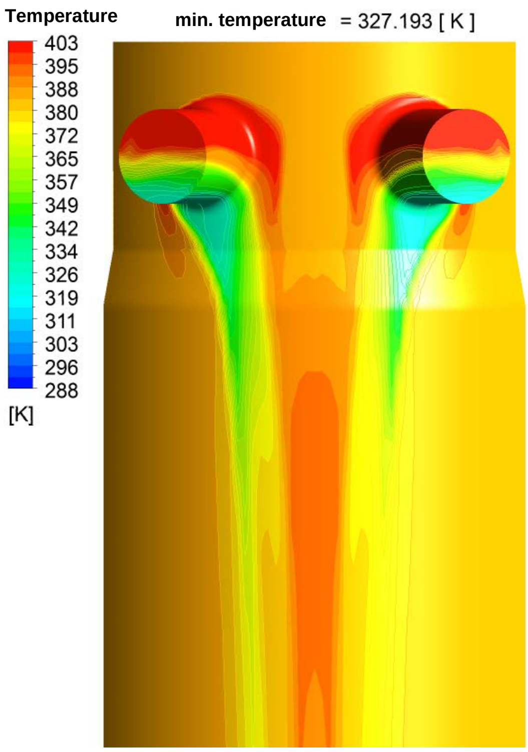
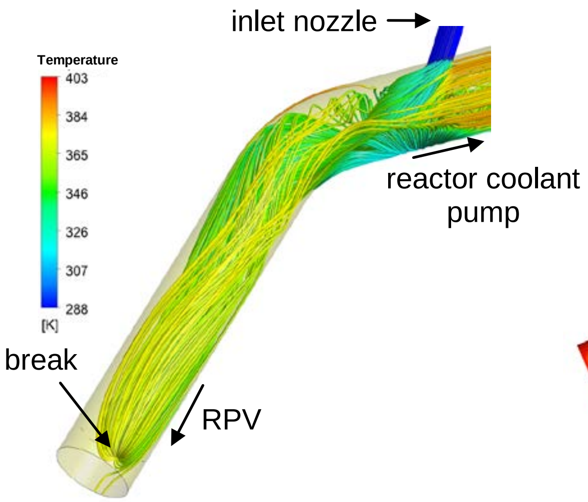


Fig. 5: Cold water streaks in the downcomer



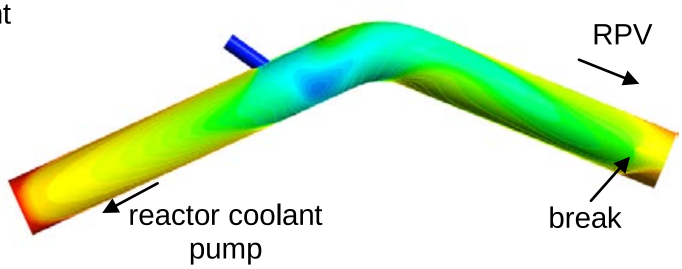


Fig. 6: Streamlines of emergency coolant

Fig. 7: Temperature distribution (pipe bottom)

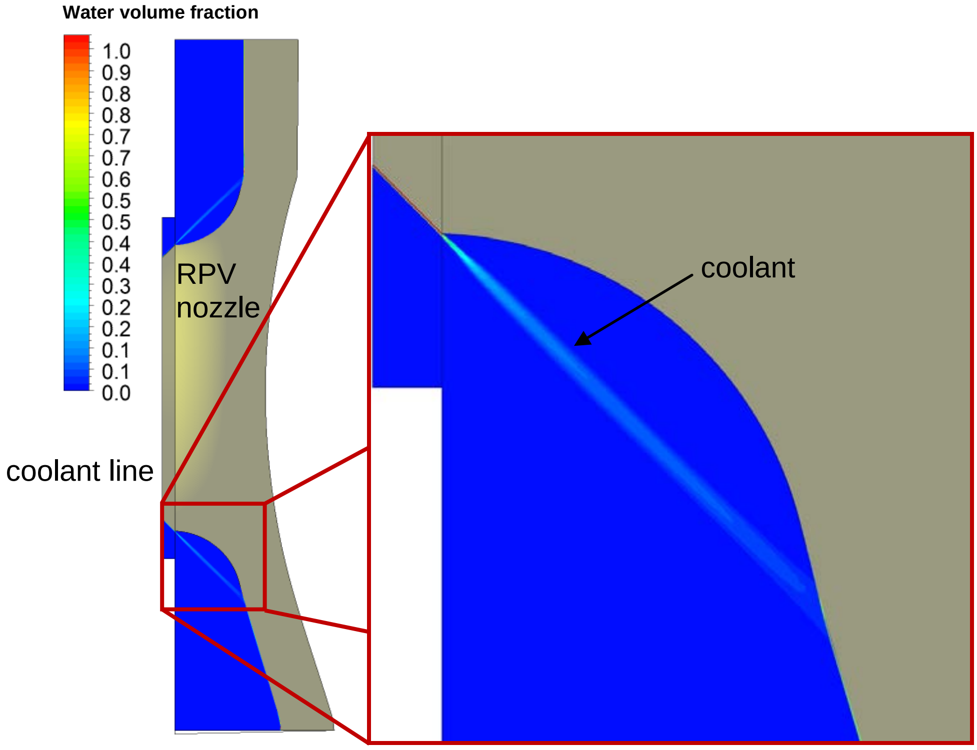


Fig. 8: Side view of the RPV inlet nozzle and

4 CONCLUSIONS

In this paper the requirements for the use of CFD analyses in regulatory practice have been derived from the German technical rules. In particular the verification and validation of the CFD code as well as the quantification of uncertainties are basic prerequisites for the computation of reliable numerical results. Because these requirements cannot be handled only by the code, the user itself is responsable for an appropriate use of CFD methods, too.

From the experience of the authors it can be stated that CFD analyses play only a secondary role in regulatory practice. This is surely caused by the high requirements and complexity of CFD analyses and the corresponding computational effort. In most cases CFD analyses only support the submitted analyses. However, there is an obvious trend to use best-estimate analyses indicated by many world-wide research activities in this field. This research will improve the knowledge base and deliver the necessary experimental data to reinforce the reliance in CFD analyses. In the end the immense progress in computing power will offer the opportunity to compute very complex flows and it will be possible to exploit the advantages of CFD methods.

5 LITERATURE

[1] ANSYS Germany GmbH
ANSYS CFX Users Manual
www.ansys-germany.com

[2] ANSYS Structral Mechanics
http://www.ansys.com/Products/Simulation+Technology/Structural+Mechanics

[3] Bundesminister des Inneren
Gesetz über die friedliche Nutzung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz)
Federal law from 15th July 1985, Bonn
Version from 08th December 2010

[4] Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
Safety Requirements for Nuclear Power Plants
from 22th November 2012
Az. RS I 5 – 13303/01

[5] Deutsches Institut für Normung
DIN EN ISO 9001:2008
Chapter 7.6: *Lenkung von Überwachungs- und Messmitteln*

[6] Gesellschaft für Anlagen- und Reaktorsicherheit mbH
Athlet User's Manual Mod. 2.0 Cycle A
GRS-P-1, Vol. 1, Rev. 3, Garching, Okt. 2003

[7] International Atomic Energy Agency
Safety Report Series No. 23
Accident Analysis for Nuclear Power Plants
Vienna 2002

[8] Nuclear Safety Standards Commission (KTA)
Safety standard 3201.2
Components of the Reactor Coolant Pressure Boundary of Light Water Reactors; Part 2: Design and Analysis
Version from June 1996

[9] Nuclear Energy Agency
Committee on the Safety of Nuclear Installations
Best Practice Guidelines for the use of CFD in Nuclear Reactor Safety Applications
NEA/CSI/R(2007)5, 15th May 2007

[10] Reactor Safety Commission (RSK)
RSK Guidelines for Pressurized Water Reactors
Version from 15th November 1996

[11] Reaktorsicherheitskommission
RSK-Empfehlung
Anforderungen an die Nachweisführung bei Kühlmittelverluststörfall-Analysen
385. RSK meeting on 20th and 21th July 2005
Version 04.2006

[12] Siemens
S-RELAP5 Code Manual
Technical Report KWU NDS1/95/E1510, Dec. 1995

[13] Studsvik Scandpower, Inc.
CASMO-5/CASMO-5M
A FUEL ASSEMBLY BURNUP PROGRAM
SSP-08/405

[14] Studsvik of America, Inc., Studsvik AB
SIMULATE-3 Methodology
Advanced Three-Dimensional Two-Group
Reactor Analysis Code
STUDSVIK/SOA-95/18

[15] T. Neuhaus, A. Schaffrath
The Pressure Surge Computer Code DYVRO mod. 3: Modelling and Validation
The 13th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-13)
Kanazawa City, Ishikawa Prefecture, Japan, September 27th - October 2nd, 2009

[16] U.S. Nuclear Regulatory Commission
Best Estimate Calculations of Emergency Core Cooling System Performance
May 1989