Reactor safety research is carried on to improve the evaluation of risks associated with the operation of light water reactors, and to diminish these risks where necessary. For this purpose, both deterministic approaches to analyze potential accident scenarios and probabilistic methods of risk and reliability analysis are used. At the present time, increasing use is made of an approach attempting to couple probabilistic analyses with a deterministic approach and, in this way, generate more precise information.

In accordance with the advanced state of the art, the consequences of severe accidents are now included in the evaluation of design basis measures. In Germany, this extension of the evaluation base was required for future facilities under the omnibus law of 1994 (amendment to the Atomic Energy Act).

Initially, accident sequences are determined mainly by what goes on in the reactor pressure vessel, the “in-vessel” phenomena. They establish the primary source term for fission products and hydrogen in the containment as well as the starting conditions, such as the temperature and composition of the melt, affecting the further course of an accident outside the pressure vessel, the “ex-vessel” phase. However, this has been dealt with in earlier contributions to this journal.

Detailed thermohydraulic mechanistic codes, such as ATHLET-CD, ICARE/CATHARE, SCDAP/RELAP5, SCDAPSIM, are used to develop in-plant accident management measures (AMM) for the primary system of a nuclear power reactor. The same codes are used to study measures seeking to improve the economic performance of existing plants with respect to their impact on plant safety.

The general purpose of all analytical codes, which is to provide reliable analyses of plant transients and accidents, can be achieved by proper modeling of important phenomena and processes and their interactions in the reactor core and the primary system. Suitable experiments are analyzed to validate and update these codes; the extrapolability to reactor conditions is verified on the basis of international comparative studies, so-called benchmarks. This contribution presents the current status of this development and validation for the design basis area as well as the beyond-design-basis regime.

In recent years, a number of advanced code systems describing thermohydraulics were coupled with three-dimensional neutron kinetics models. Such coupled code systems are being used increasingly in research, industry, and by regulatory authorities to study complex accidents (such as reactivity transients, deboration transients or subcooling transients) with pronounced spatial differences in power distribution inside the core. In this way, safety-related local parameters can be determined in the light of various factors, such as enrichment, burnup, etc., and the safety margins of core configurations can be assessed with greater precision.

In addition to real data from power reactors, also numerical benchmarks are used for further development and validation of these coupled code systems. These include the international benchmarks of OECD/NEA and of the Code Assessment and Maintenance Program (CAMP) of USNRC.

The NUKLEAR Program of the Karlsruhe Research Center, in a cooperative venture with Framatom/ANP, Erlangen, successfully participated in the international benchmarking exercise on the main steam line break with their RELAP5/PANBOX code system [1]. The progress achieved is apparent from a comparison of the reactor power (Fig. 1) calculated by point kinetics and 3D kinetics. The important feature is the lower power rise after a reactor scram as determined by 3D kinetics as compared to the level determined by point kinetics (up to 600 MW).

In addition, more detailed information is obtained about the change in time of axial and radial power distributions. The power distortion visible in Fig. 2 is due to asymmetrical core cooling and assumed blockage of an absorber rod bundle (“stuck rod”). These pronounced spatial modifications highlight the problems inherent in the point kinetics approach which is based on the as-
sumption of a constant axial power profile function.

The RELAP5/PARCS code system is currently being qualified within the framework of the international VVER-1000 coolant transient (V1000-CT) benchmarking exercise. Experimental data measured in the Kozloduy nuclear power station are available for this purpose.

**Adjoint Sensitivity Analysis Procedure**

The Adjoint Sensitivity Analysis Procedure (ASAP) [2] is a method of determining the sensitivities of the results of thermohydraulic models as a function of all systems parameters, α, to a systems result, R.

\[
\left(\text{Definition: } \frac{\partial R}{\partial \alpha} \cdot \frac{\alpha}{R}\right)
\]

The relative sensitivities allow the importance of the respective parameter to the overall systems result to be determined.

The implementation of this method in RELAP5/MOD3.2 has been completed successfully and tested on the basis of the QUENCH-04 experiment [3]. Figure 3 shows the example of time-dependent behavior of the eight highest relative sensitivities of the temperature of the outside wall of the heated rod at a level of 1.3 m.

The parameters shown have this physical significance: nominal power factor (α1), nominal power up to 121 s (α2), nominal power up to 2088.6 s (α3), nominal pow-

![Fig. 1: Comparison of the reactor power computed by point kinetics and 3D kinetics for the TMI-1 reactor for the steam line break scenario.](image1)

![Fig. 2.: Radial distribution of the power profile as computed by the 3D kinetic model.](image2)
er up to 2103 s (\(\alpha_4\)), nominal multifactor of the internal source (\(\alpha_5\)), nominal surface temperature at 1.3 m (\(\alpha_6\)), nominal volumetric heat capacity of the ZrO\(_2\) pellet (\(\alpha_7\)), and nominal volumetric heat capacity of zircaloy (\(\alpha_8\)).

The sensitivities of the \(\alpha_1\), \(\alpha_4\), \(\alpha_3\), and \(\alpha_5\) parameters are positive, i.e. their values increase with time. The sensitivities of parameters \(\alpha_7\) and \(\alpha_8\) are negative, i.e. their values decrease with time. Above all, the relative sensitivity of the temperature of the outside wall after 800 s reaches approx. 56 % with respect to the nominal power factor, i.e., a 10 % change in the nominal power factor results in a 5.6 % change in the temperature of the outside wall. The other relative sensitivities are comparatively low.

**Flooding a Superheated Reactor Core**

Improving predictions of the response of a superheated reactor core to a flooding event is a problem under study worldwide. One contribution to this topic by the Nuclear Safety Research Program (NUCLEAR) of the Karlsruhe Research Center was the QUENCH-06 experiment [3], which OECD/NEA selected as the International Standard Problem (ISP) No. 45, and which was run under the responsibility of the Research Center. It includes studies of core behavior in nuclear power plants during heating and delayed flooding in an assumed loss-of-coolant accident.

In order to be able to assess the status of core meltdown codes for simulating core heating and quenching with water at temperatures above 2000 K, the so-called blind phase of the ISP was given only the initial conditions and boundary conditions necessary for recalculating the experiment; no other experimental details were provided. 21 organizations from 15 countries participated in this phase with eight different code systems (ATHLET-CD, ICARE/CATHARE, IMPACT/SAMPS0N, GENFLO, MAAP, MELCOR, SCDAPSIM, SCDAP-3D), and IRS was involved with a version of its own, SCDAP/RELAP5 mod3.2.irs (S/R5).

Up to and including the second heating phase (\(t = 7\) ...s, Fig. 4), most of the results did not differ greatly other than as a consequence of obvious user errors; this allowed a “main field” of results to be defined. For the quenching phase (\(t > 7\) ...s), modeling of thermohydraulics was found to be insufficient: some participants were not able to model the cooling rates observed experimentally, while others had to use a very fine grid to balance out inadequacies in nodalization of the computer code.

In the experiment, a sufficiently thick oxide layer prevented failure of the cladding tube below \(~ 2200\) K and thus the release of molten metal. In most computer codes, this behavior was described by the standard oxidation models, unless spalling of the oxide layer is assumed arbitrarily. In the main field, variation of the \(\text{H}_2\) masses released as calculated increased from approx. \(\pm 15\)% before flooding to approx. \(\pm 40\)% after flooding (Fig. 5).
The outliers among the calculated temperatures and hydrogen masses released are based on the assumption of the protective oxide layer spalling. In the main field of results, most participants correctly calculated that there would be no destruction of the bundle. Detailed examination also showed that the codes still have problems in correctly calculating the initial conditions in the bundle at the onset of flooding.

Another surprising result is this: The energy balance needs to be checked carefully before interpreting any findings. Lack of experience among the code users, and difficulties in adequately modeling the QUENCH facility, were the main causes of major deviations.

The results of the open phase show that the codes are able to analyze the experiment satisfactorily by parameter fitting. Some participants corrected errors or improved modeling of the code. Variance in the findings, e.g. in
the total H₂ mass calculated, was clearly reduced (Fig. 5).

The validation of ASTEC V1 on the basis of QUENCH-06, which was begun last year, indicated deficits which are to be repaired in SARNET within the 6th EU Framework Program.

The database about core behavior in delayed flooding was clearly expanded in the QUENCH program, but there is still need to examine conditions at low flooding rates and extensive damage conditions, such as a local bed of debris.

**Final Phase of a Severe Core Meltdown Accident**

The experimental Phebus-FP (fission product) Program is carried out by international partners at Cadarache, France, for prototype studies of fuel rod behavior up to the late phase of a core meltdown accident.

The Phebus-FP series of experiments serve to study the thermo-mechanical and physico-chemical phenomena in an LWR core meltdown accident, especially the release of fission products, and make these findings available for code validation.

The series comprises six experiments, four of which have already been conducted: three experiments with fuel elements containing 20 fuel rods each and one central absorber rod; one experiment with a predefined bed of fuel and oxidized cladding tube fragments [6].

In the FPT1 Phebus experiment, the fuel element (approx. 1 m long) consisted of 18 irradiated fuel rods (burnup ~ 23 GWd/tU) and two fresh fuel rods as well as one central control rod with silver-indium-cadmium as the absorber material, a stainless steel cladding tube, and a zircaloy guide tube. Heating of the fuel rod bundle was controlled by the rising nuclear power and the steam mass flow (0.5 – 2 g/s); an excess supply of oxygen was provided for zircaloy oxidation.

The experiment has been analyzed and interpreted in detail as ISP-46 of OECD. IRS participated in this effort by recalculating...
the first phase of the experiment, i.e. fuel rod failure and fuel rod meltdown, melt relocation and molten pool formation, hydrogen generation due to zirconium oxidation [7]. The ICARE2 and SCDAP/RELAP5 mod 3.2.irs (S/R5) computer codes were used for this purpose.

Figure 6 shows a radiograph of the final state of the bundle with a central region largely free from fuel and a molten pool at the bottom end. Figure 7 shows that the calculated fuel rod temperatures are in good agreement, both qualitatively and quantitatively, with the systemically induced lower levels of the ultrasonic thermocouples; in the best-estimate case at the axial position of 0.3 m, ICARE2 calculates fuel superheating to approx. 3000 K, which is due to a local blockage caused by dislocated fuel. One parameter very important in assessing the consequences of an accident is the cumulated volume of hydrogen released as a result of zirconium oxidation. The calculated hydrogen mass (Fig. 8) is within the uncertainty band of the measured results of approx. 20%.

These and other analyses of the Phebus-FP experiments demonstrate that both ICARE2 and SCDAP/RELAP5 are able to describe with good accuracy the key phenomena of an LWR core meltdown accident up to fuel dislocation and molten pool formation. However, the experiments indicate that massive core degradation can occur even some 250 K below the melting temperature of (Zr,U)O₂, with burnup and, thus, the fuel structure exerting a major influence. In both computer codes, these influences of burnup on the onset and development of fuel dislocation are taken into account only in a greatly simplified way, which

![Fig. 6: Radiograph of the FPT1 test section [6].](image)

![Fig. 7: Fuel rod temperatures at 0.3 m. Red: data measured by ultrasonic thermometers; blue/green curves: ICARE2 and S/R5 results, respectively.](image)
means that quantitative statements about the origin and propagation of the resultant molten pool in the late accident phase are still subject to major uncertainties.

Future work will concentrate on qualifying the TRACE/PARCS code system, the successor to the RELAP5/PARCS system, for application to light water reactors and ADS systems.

The new versions of the integral French-German ASTEC accident code will continue to be validated on the basis of the QUENCH experiments.

Preparation of the two QUENCH experiments in the LACOMERA EU program will be backed by advance calculations and analyses. As for QUENCH, analytical support will be continued for the integral experiments, such as Phebus STLOC (to study the influence of burnup).

Experimental findings about core reflooding as an accident measure will be systematized with respect to the parameters influencing them, such as core degradation state, flooding rate, systems pressure, feeding location, burnup, etc., and prepared for probabilistic applications.

**Outlook**

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**Literature**


[3] QUENCH code, see contribution to this publication


