STATUS OF THERMOHYDRAULIC RESEARCH IN NUCLEAR SAFETY AND NEW CHALLENGES

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ABSTRACT

Thermohydraulic phenomena, to be expected with design basis accidents, are well understood today and there is also a fair knowledge of fluiddynamics, which would be encountered with severe accidents, being affected with small or moderate core degradation.

Additional research efforts are needed for reliable and precise descriptions of thermo- and fluiddynamic phenomena and effects with highly disintegrated core and with hazardous conditions in the containment. From the design and development of new Advanced Watercooled Reactors new and interesting challenges originate for thermohydraulic research in nuclear safety. An important task for the future will be to sustain an adequate level of nuclear safety research and to preserve the know-how, gained over the last decades in this field.

I. INTRODUCTION

Due to the fact, that world-wide almost all nuclear power stations are using water as working fluid, thermohydraulic research for nuclear reactors is mostly concerned with two-phase flow phenomena. In the very early period of nuclear power development, fluiddynamic research was concentrated on questions arising from normal operation. Such questions were for example void fraction with subcooled and saturated boiling, sub-channel mixing, pressure drop or critical heat flux. Starting mid of the sixties, accident analysis became more and more dominant.

Today more than 90% of thermohydraulic research for nuclear reactors belong to the field of nuclear safety. The last barrier against radioactive impact to the environment is the containment and therefore nuclear safety research concentrates also to fluiddynamic and thermodynamic phenomena in the containment atmosphere, like single-phase natural convection or hydrogen combustion after a severe accident.

Due to the major role of nuclear safety problems in thermohydraulic research, the explanations of this paper are restricted to nuclear safety questions and issues.

II. HISTORICAL EVOLUTION OF THERMOHYDRAULIC RESEARCH IN NUCLEAR SAFETY

In the sixties and at the beginning of the seventies, there was a world-wide optimism, that the design of emergency installations and the planning of accident procedures can be deduced from a scenario arising from a so-called "Maximum Credible Accident", which was usually postulated as the double ended break of a main coolant pipe. Lateron a few more scenarios were added to these deliberations. This "limiting" scenario, which should cover all "credible" accidents, was called "Design Basis Accident".

The first risk study, carried out by Rassmussen and his colleagues, pointed out, that there are several other scenarios, which have a higher probability and may lead to more severe consequences, than the double ended break. One concern was the small leak with only small energy transport out of the primary system, which may result in a partial core melt. This scenario unfortunately became reality in the Three Mile Island Accident, 1979.

Other risk studies followed and their findings resulted in a world wide research on scenarios, assuming beyond design severe accidents. The accident in the RBMK-reactor of Tschernobyl, 1986, enforced this movement in spite of the fact, that the nuclear and thermohydraulic phenomena, which occurred in Tschernobyl, physically could not happen in a Pressurised or Boiling Water Reactor of western design. End of the eighties and beginning of the nineties probabilistic studies on risk assessment were strongly enforced in many western countries. This finally led to the consequences, that strategies were derived to mitigate the consequences of severe accidents by accident management procedures. Finally, shaping of environmental ideas in the public encouraged the vendors to design new reactor concepts, equipped with - as much as possible - passive emergency systems to assure inherent safety features to a great extend. Passive in this connection means, that for example the emergency core cooling is guaranteed by gravity force only and there is no need of an active pump. In buoyancy driven flows, low velocities exist and the fluiddynamic system is very sensitive to small changes in pressure drop. These features will give new challenges to fluiddynamic research, both in theoretical code development and in experimental activities.

Following the changing philosophy in licensing, one can also observe a varying basis for accident analysis. As long as design basis accidents were taken in account, so called conservative assumptions in analysing scenarios and consequences of nuclear accidents were quite appropriate for proving the safe design of the plant. Risk studies and especially the planning of accident management against severe accidents demand a best estimate analysis, which means, that the fluiddynamic situations to be expected during accident sequences, have to be known as good and realistic as possible. This again resulted in the consequence, that the physics had and have to be studied much more precisely and in detail. From these questions arose with respect to fluiddynamic modelling, to scaling and to a proper description of physical effects and phenomena in the codes. This movement was enforced by special questions, connected with designing passive emergency cooling systems.

In future the fluiddynamics probably must be formulated in a much more sophisticated way, as it is done up to now in system codes, based on lumped parameters. There may be a need to link modern fluiddynamic codes with the system codes, used up to now in nuclear safety analysis. New experiments must be performed to provide the necessary physical input into these combined code systems.

III. FLUIDDYNAMICS IN PRIMARY SYSTEMS

In 1996 the OECD edited the SESAR/FU report /1/, in which it is stated, that a common technical position exists in that "extensive knowledge is available in the field of transients and major system computer codes have achieved a high degree of maturity". Codes like

APROS (Finland)
ATHLET (Germany)
CATHARE (France)
CATHENA (Canada)
RELAP 5 (U.S.A.)
TRAC (U.S.A.)

are available to describe the system behaviour in the primary loops during incident and accident situations. As with all codes dealing with thermo- and fluiddynamic problems, also in these codes one can distinguish two sets of equations, namely those which are formulating the conservative laws (balance equations) and others, which are modelling the physical phenomena (constitution equations). In early years, the balance equations were not formulated for each phase - water and steam - separately and the two phase system could be formulated rather simply, just by using equations for the void fraction or the slip between the phases and empirical correlations for the pressure drop. Advanced codes formulate the conservation laws for each phase separately, ending up with 6 balance equations, which requires much more detailed knowledge about phase-interface interactions and transport processes for momentum mass and energy. It would blow up the scope of this paper too much to discuss only a few of the phase-interface phenomena, being of interest in nuclear thermohydraulic research. There are world-wide many experimental facilities doing research in so called separate effect tests.

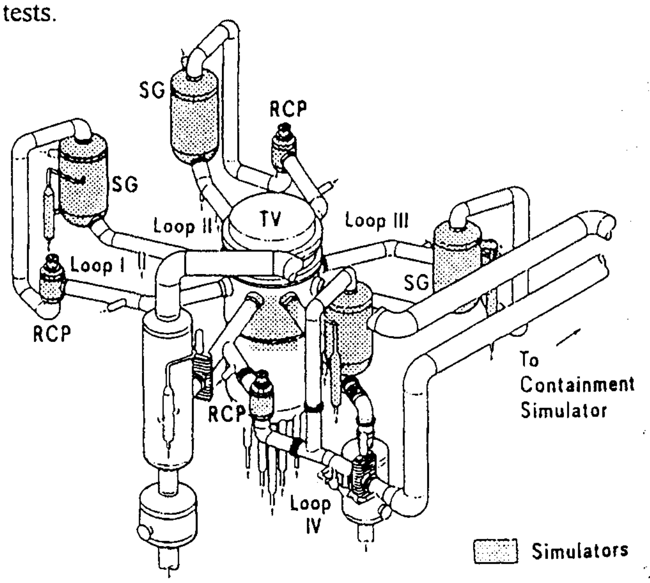

Fig. 1: Upper Plenum Test Facility (UPTF)

Another problem of major concern is the geometrical scaling in thermohydraulic research. Nuclear safety experimental facilities usually have the full height of the primary system of a water cooled reactor, but due to economical reasons, which are easily understandable, their cross-sections are smaller by 2 or 3 orders of magnitude compared to the reactor. So they represent the correct buoyancy forces, but they do not have the same ratio of friction and momentum forces as in the reactor and - what is especially important - they represent only one-dimensional fluiddynamics. Three-dimensional effects are very relevant in the annular downcomer of a pressurised water reactor during accumulator injection after a loss of coolant accident or in the hot leg during decay heat removal via boiler-condenser mode. Both examples will be briefly described here.These multidimensional and macro-scale effects were studied in the German Upper Plenum Test Facility, which is - as Fig. 1 demonstrates - a full sized copy of 1300 MW_e of the primary loops of a pressurised water reactor, however with simplified components and only working up to a pressure of 20 bar. There are 4 loops, one of them representing the broken loop, in which the leak can be of any size and can be situated in the cold or in the hot leg. The full sized pressure vessel contains a dummy-core, which means, that instead of boiling evaporation by electrical heating, vapour and also water can be blown through a cluster of fuel elements, representing the full cross-section of a reactor core and so imitating the fluiddynamics incorporated with decay heat removal.

Also the dummies of the main circulation pumps and of the steam generators are passive devices. They are mainly imitating the pressure drop, which the real component would have, in case of emergency core cooling and decay heat removal.

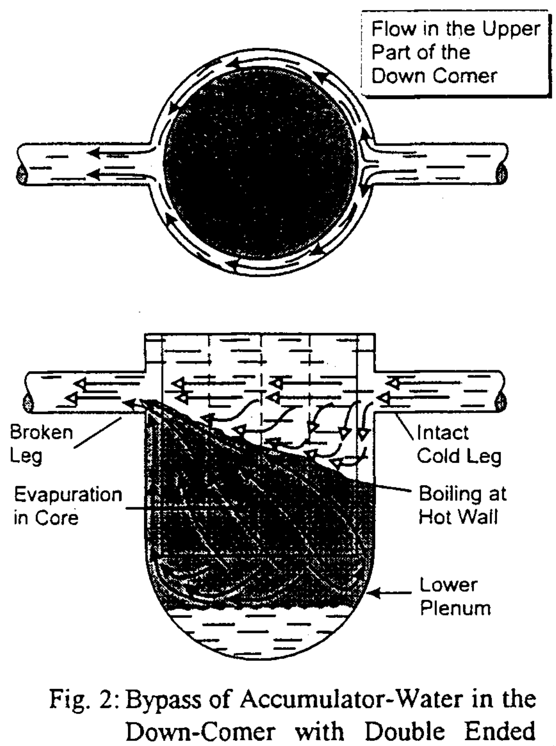

Fig. 2: Bypass of Accumulator-Water in the Down-Comer with Double Ended Cold Leg Break

Experiments in this facility were carried out during the international 2D/3D Research Program /2/, in which Japan, U.S.A. and Germany closely co-operated. Following this co-operation, the German Government sponsored extensive research in this facility for studying decay heat removal with degraded emergency core cooling systems and via reflux-boiler-condenser mode. This program was mainly devoted to prove the efficiency of accident management measures during severe accident scenarios /3/. This program was terminated end of last year.

In case of a large leak in the cold leg, emergency core cooling water early injected by the accumulators, may flow around the annulus of the downcomer to the broken position, due to strong pressure forces, as demonstrated in Fig. 2. In addition, steam produced by evaporation in the lower plenum and in the core, flowing up to the leak also, prevents the emergency cooling water from flowing down. So most of the water, injected by the accumulators may not reach the lower plenum and cannot contribute to the cooling of the core.

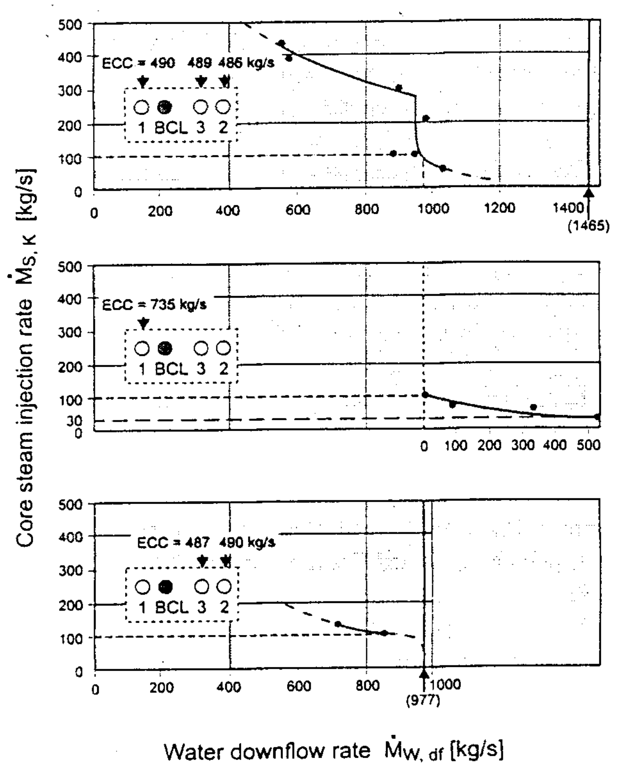

Fig. 3: Effectiveness of Accumulator Injection via Various Cold Legs (Upper Diagram: Injection through 3 Cold Legs; Middle Diagram: Injection through 1 and Lower Diagram: Injection through 2 Cold Legs)

In detail this situation however is a function of the positions of the broken leg and the injecting legs and of the steam produced in the core. Fig. 3 gives an imagination of this dependency between ECC-position and steam production in the core on one side and the water downflow rate on the other.

In the upper diagram of Fig. 3, water from the accumulators is injected through 3 cold legs into the downcomer. Even with high steam upflow rate (300 - 400 kg/s) enough downflowing water (600 - 800 kg/s) is reaching the core for fast quenching and safe heat removal. The total amount of water injected into each of the 3 cold legs is 488 kg/s. In case of injecting emergency cooling water through 2 cold legs only, which however are situated on the opposite side of the broken leg, still enough water is reaching the lower plenum, as the lower diagram in Fig. 3 demonstrates. If only 1 emergency core cooling system would work and if this in addition is situated just next to the broken leg, the cooling of the core may become endangered, because only with very low steam production considerable amount of accumulator-water can reach the core, as one can see from the middle diagram of Fig. 3.

There is no general scaling law - for example by the aid of dimensionalless numbers - which could model these phenomena and small scale experiments do not show this geometrical effect. The situation differs from reactor to reactor, depending on the position of the cold leg nozzles around the annulus of the downcomer.

One could imagine, that the fluiddynamic situation in the hot leg under counter-current flow is as complicated as discussed before for the downcomer. Counter-current flow in the hot leg would occur during a severe accident, when in case of a station black-out, the decay heat would have to be removed from the core via boiler-condenser-mode, which means, that the steam, produced in the core flows via the hot leg to the steam generator, is condensing there and the water has to go back against the steam flow, driven by gravity force only. In case that the shear-stress between the phases is larger than the gravity force, the water would be prevented from downflowing and a situation would occur, which is called "counter-current flow limitation" (CCFL).

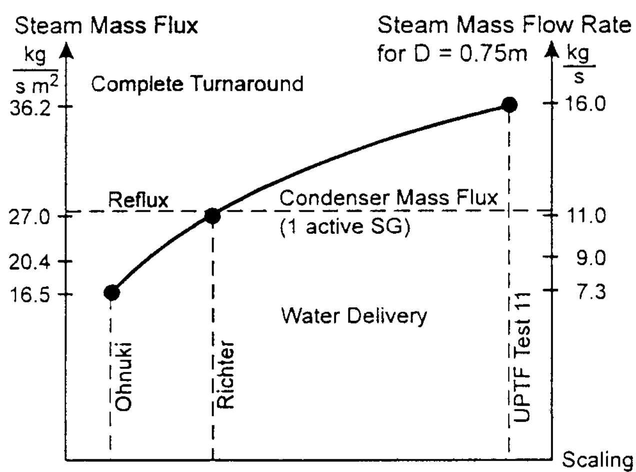

Fig. 4: Counter Current Flow Limitation in the Hot Leg for Boiler Condenser as Found in Various Experimental Facilities

Also for this cooling condition, the question arises: how do the geometrical conditions - the design of the hot leg, its bend and the scale, the diameter of the pipe - influence CCFL.

There are several studies on CCFL in the hot leg with reflux boiler condenser mode. Here the results of three studies, namely by Ohnuki /4/, by Richter /5/ and the experimental results in the large scale test facility UPTF wil be briefly compared. In Fig. 4 the steam mass flux (left side of the scale) and the total steam mass flow rate (right side of the scale) are plotted versus the size or respectively the scaling factor of the test facility. Ohnuki's test facility was scaled down by a factor of 10 and Richter's by a factor of 3 with respect to the reactor dimensions (0,75 m). At first glance there seems to be a clear influence of the scaling. What is not seen in this figure is the fact, that the design of the hot leg and especially of the bend was different in all three cases. This configuration of the bend has the strongest influence on the CCFL and the scaling effect in the range between 0,07 and 0,75 m is by far not as strong.

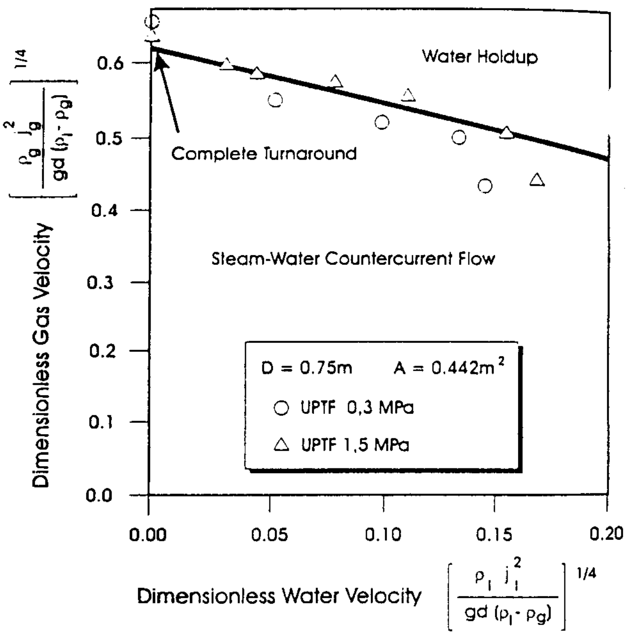

Fig. 5: Counter Current Flow Limitation for Boiler Condenser Mode in the Hot Leg, as Measured in UPTF

Pipe diameter d and also pressure can be sufficiently well scaled, if the experimental data are plotted with the help of dimensionalless velocities, as done in Fig. 5 on the basis of the Wallis number

$$\left[\frac{\rho_g\, j_g^{\,2}}{gd\left(\rho_l-\rho_g\right)}\right]^{1/4} \quad or \quad \left[\frac{\rho_l\, j_l^{\,2}}{gd\left(\rho_l-\rho_g\right)}\right]^{1/4}$$

The boundary line separating steam water counter-current flow and water hold-up (CCFL) is the same for various pressures.

IV. FLUIDDYNAMICS WITH PARTIALLY DISINTEGRATED CORE

Situations with partial disintegrated core open a wide scenario of phenomena, both, in the primary system and outside in the containment atmosphere. There are several fields of interest resulting from effects, which may endanger the containment. Such phenomena are

hydrogen formation, distribution and combustion,
steam explosions,
direct containment heating,
pressure rise in the containment and
core-concrete interaction.

With respect to the primary system, there are activities in the United States and also at other places to prevent the pressure vessel from failing by cooling from outside. The idea is, that the heat transport from the molten debris inside the vessel through the vessel wall and to a water pool outside is good enough to avoid, that the wall of the vessel is molten from inside to such an extent, that it would fail be mechanical loads. Object of thermohydraulic studies in this connection is the free convection of the melt inside the lower plenum of the pressure vessel. But also the heat transfer, due to boiling at the outside surface of the wall of the pressure vessel was under consideration /6/.

There are several lessons, which could be learned from the TMI-accident. One came out after the core debris was removed from the lower plenum. In spite of the prophecies of doom, the wall of the lower plenum was not attacked by the molten debris to such an extent, that it was just on a situation to fail, but in contrary, only a spot, having a diameter of approximately 1 m, showed annealing colours. The whole surface of the vessel wall in the lower plenum remained smooth, even though it was housing 20 tons of core debris. The reason for this probably was, that there remained water in the lower plenum before the core started to melt and that also the core granulate was always covered by water.

This gave the hint to study heat removal out of a "pebble bed" of granulated core debris being situated in the lower plenum. An example for a test facility, in which such studies are performed, is shown in Fig. 6. This facility is operated at the Technical University of Munich. The main component of the test facility is a scaled down pressure vessel, containing the test objects, which is designed up to a pressure of 100 bar. Auxiliary components, like a feed water tank and a storage tank, can supply cooling liquid into the pressure vessel. The steam generated there can flow to a condenser to be liquefied again. Tests are performed with water and also with the modelling fluid R134a. The interior of the pressure vessel shows Fig. 7. The heating of the debris substitutes is carried out by an induction-coil and cooling liquid can be added from the top to the debris. There are being studied two kinds of debris-substitutes. The first one consists of a pebble bed of steel balls, each of them having 4 mm diameter and the other one is a mat made of sintered and porous material. Both are heated by induced electrical current.

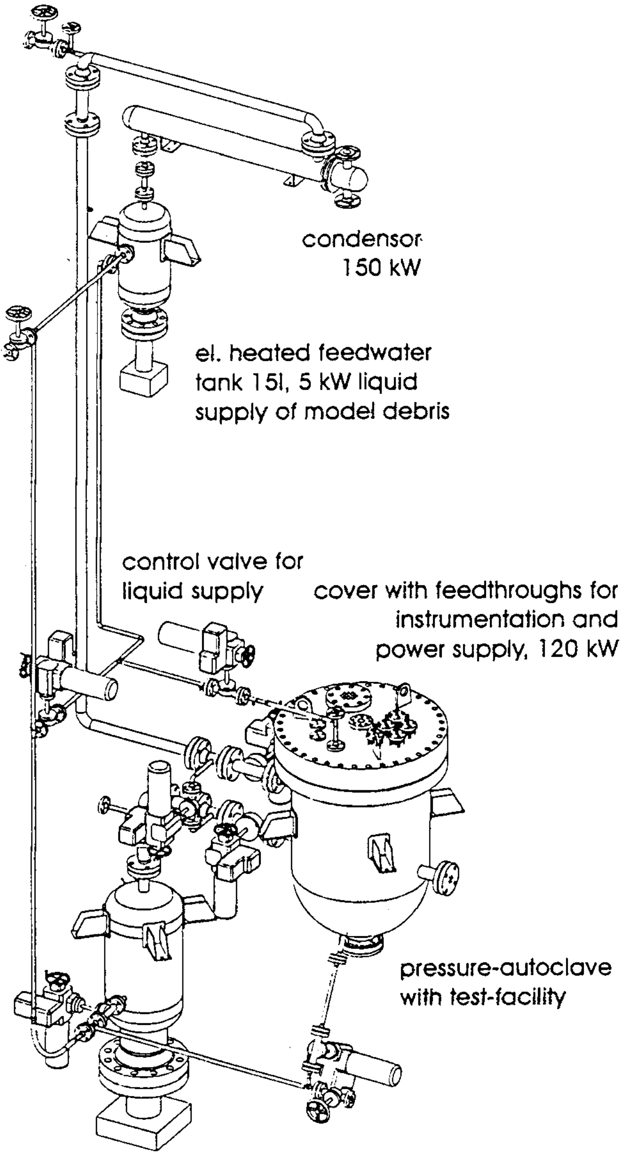

Fig. 6: Test Facility for Studying Coolability of Granulated Core Debris in the Lower Plenum

As shown in Fig. 7, these debris substitutes are placed in a receptacle, the walls of which can be made from steel or from glass. Glass walls have the benefit, that both, the temperature of the bottom of the substitute and the movement of the two-phase mixture there, can be observed by optical methods. The temperature is measured by a infrared-thermo-camera and the two-phase mixture by a high-speed movie camera.

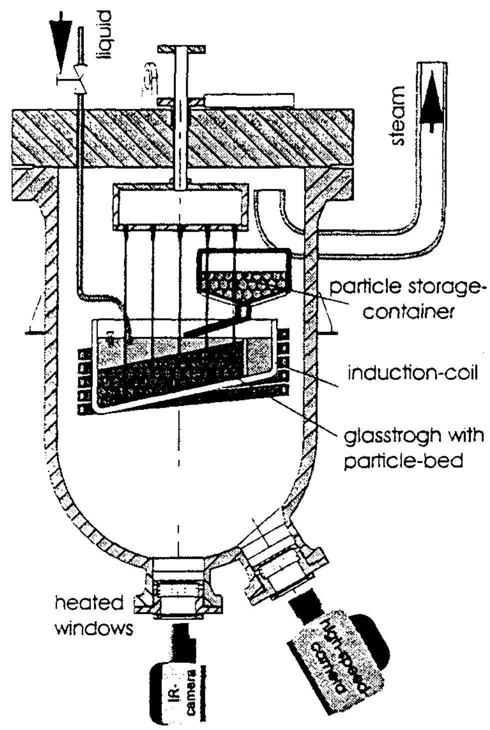

Fig. 7: Arrangement for Studying Fluid Dynamics and Heat Transfer in a Granulated Core Debris

Inside the pebble bed and the sintered mat, two kinds of void meters are distributed, one working on a capacitive basis and the other one with optical probes, using light deflection as demonstrated in Fig. 8. The liquid level or the swell level respectively can be varied in the receptacle by adding more or less fluid from the top. Also the height of the pebble bed can be varied during the experiment.

In preliminary tests, it could be observed, that also beyond the Leidenfrost temperature of the steel balls, cooling is achieved by strongly pulsating two-phase flow in the vessel up to a certain limit of volumetric heat production. This limit is a function of the pulsating period, especially of the duration of the dry period in the vessel. This duration again depends on the length of the path, the cooling liquid has to go down to the lowest layer of the steel balls after explosive like evaporation. It is also a function of the heat flux density and of the gravity force (height of the liquid layer) above the pebble bed.

Another interesting field of research, concerning severe accidents is the behaviour of hydrogen in the containment and there especially the flame acceleration and - with higher hydrogen concentrations - the transition from deflagration to detonation (DDT). Only at concentrations above 11%, hydrogen DDT is likely in a dry containment atmosphere.

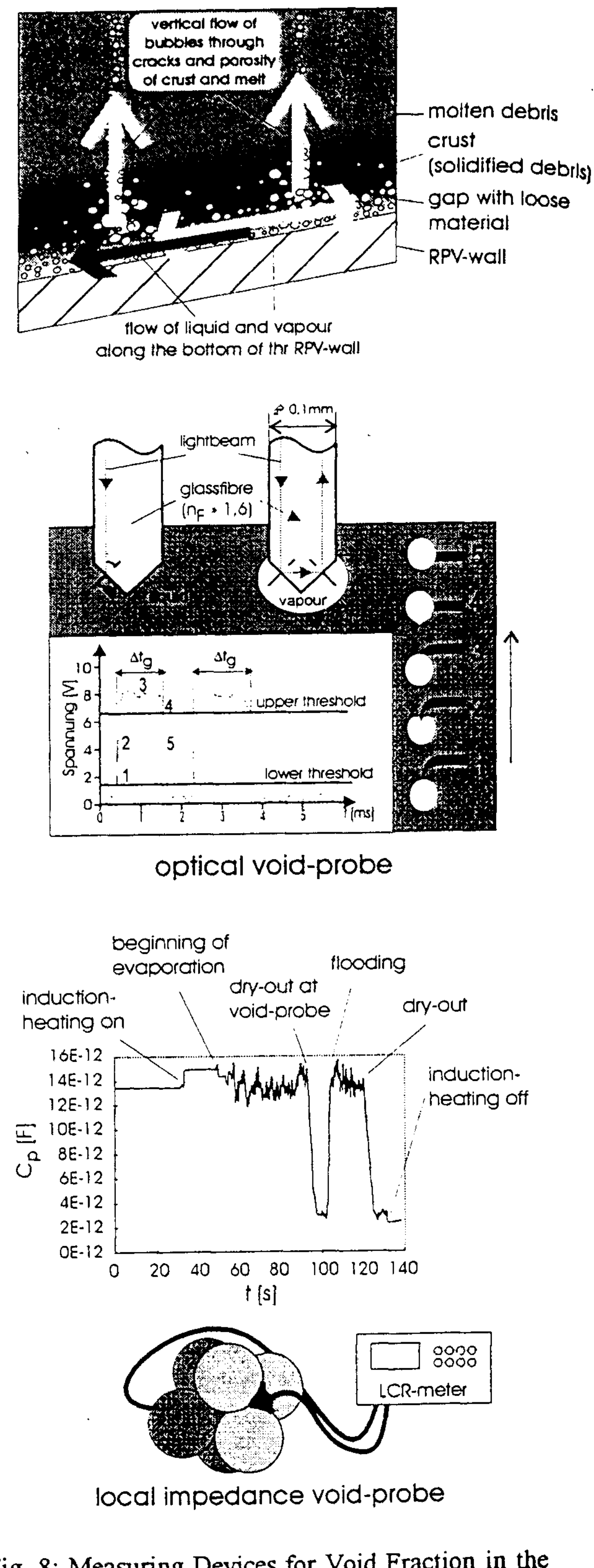

Fig. 8: Measuring Devices for Void Fraction in the "Pebble Bed" Simulating Granulated Core Debris

If there is steam in the containment - wet atmosphere - due to a blowdown of the primary system, the hydrogen concentration can be higher before a DDT can occur. Finally at very high steam content in the containment - above 35% - DDT is completely surpressed even at hydrogen concentrations, which would have to be expected in a very severe accident, if all zirkonium would have reacted with water and if even the steel in the concrete would have been seriously attacked. With an open primary system, there is always steam in the containment atmosphere. So detonation can be excluded, even after a very severe accident, according to the author's opinion.

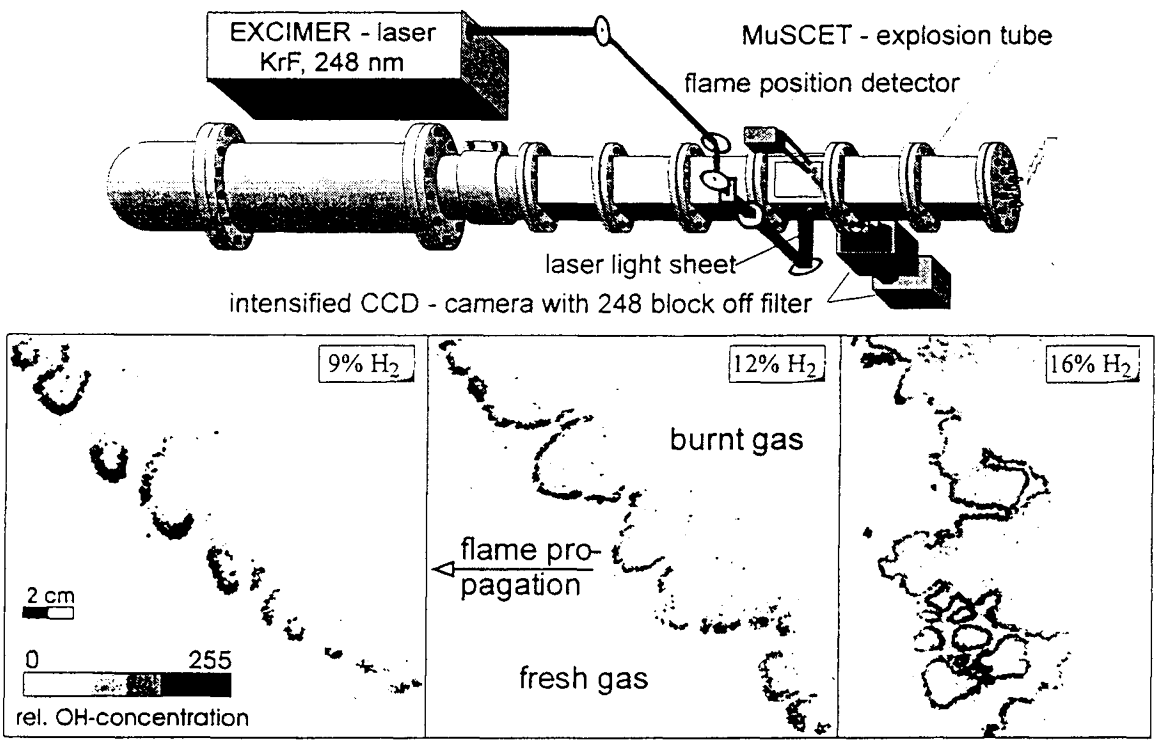

Fig. 9: Flame development with various H_2 concentrations behind obstacles (dry atmosphere)

However, flame acceleration can produce considerable pressure pulses, resulting from violent deflagrative combustion. Therefore it is of interest to know which flame velocities have to be expected in the containment.

It is well known from the literature, that a flame is accelerated by turbulence. This turbulence is produced by the flame itself, resulting from the expansion of the hot burned gas when the flame front approaches an obstacle or when it burns through an orifice into another room also filled with combustible mixture.

Measurements showed, that obstacles like pillars or grids have different effect on flame acceleration than orifices producing a jet of hot gas and highly reactive radicals. Flame acceleration behind obstacles and through orifices was measured at the Technical University of Munich /7, 8/. An explosion tube, as sketched in Fig. 9, was used to study flame development and flame acceleration behind obstacles. The flame front was monitored by laser-induced fluorescence, using an EXCIMER-laser, which emitted a light sheet through the cross-section of the explosion tube at a position when the flame had passed the obstacle. The light sheet produced a fluorescence of the OH-radicals in the flame.

The flame front is strongly dependant on the hydrogen concentration in the unburned mixture, as Fig. 9 mediates. With low hydrogen concentrations, the turbulence can produce quenching effects, which deteriorates the combustion process. With higher hydrogen concentrations, the obstacles produce a very wrinkled flame with pockets of unburned gas within the flame front and this may produce local pressure waves, finally leading from a deflagration to a detonation.

The situation is somewhat different, if we look to the flame development behind an orifice where the ignition is initiated by a cloud of burned gas and radicals jetting through the orifice. In this case, hydrogen mixtures have a very special behaviour compared to mixtures of hydrocarbons, as demonstrated in Fig. 10. The ignition and flame development in a chamber behind an obstacle, is shown there for hydrogen and for methan and the flame propagation was made visible by the use of high-speed-Schlieren-photographs. The concentration for each of the combustibles was chosen in such a way, that both had identical laminar flame velocity. Comparing both series of Schlieren photographs, one realises, that the hydrogen flame front is strongly wrinkled, as we learned from Fig. 9, whereas the methan flame front is smooth. Both jets penetrate the second chamber to the opposite wall. However, the flames, induced from these jets, are very different. The jet of the hydrogen flame ignites the combustible mixture in the second chamber explosive like, whilst methan burns slowly there, in spite of the high turbulence produced by the jet.

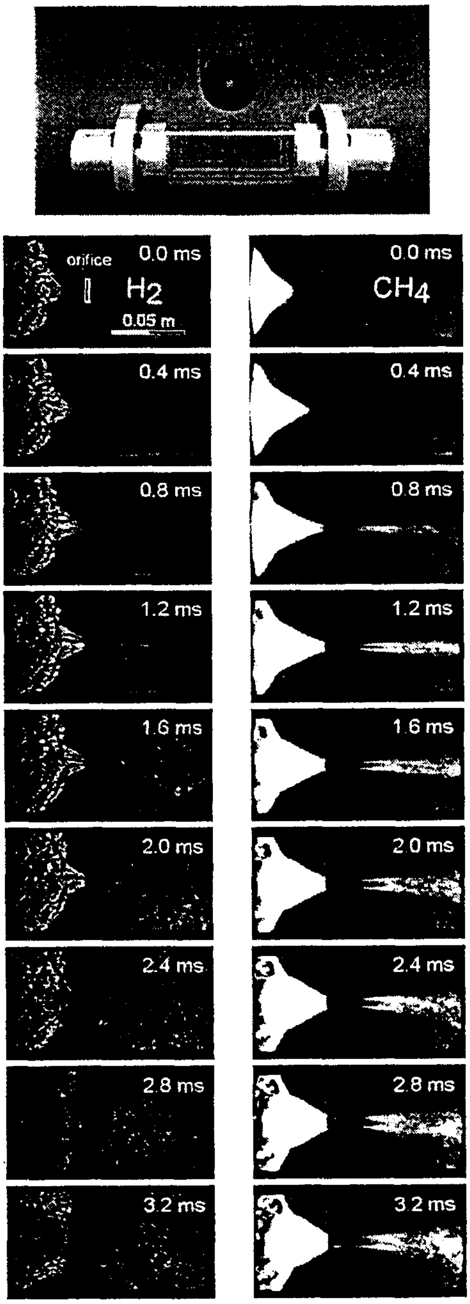

Fig. 10: Flame formation in front and after an orifice with H_2 and CH_4 as combustible.

The igniting phenomena of the jet with hydrogen as combustible are shown in Fig. 11 a little more in detail. The jet is disintegrated after a certain distance from the orifice only and the vortices, forming there, are creating distributed igniting spots, resulting in a violent combustion.

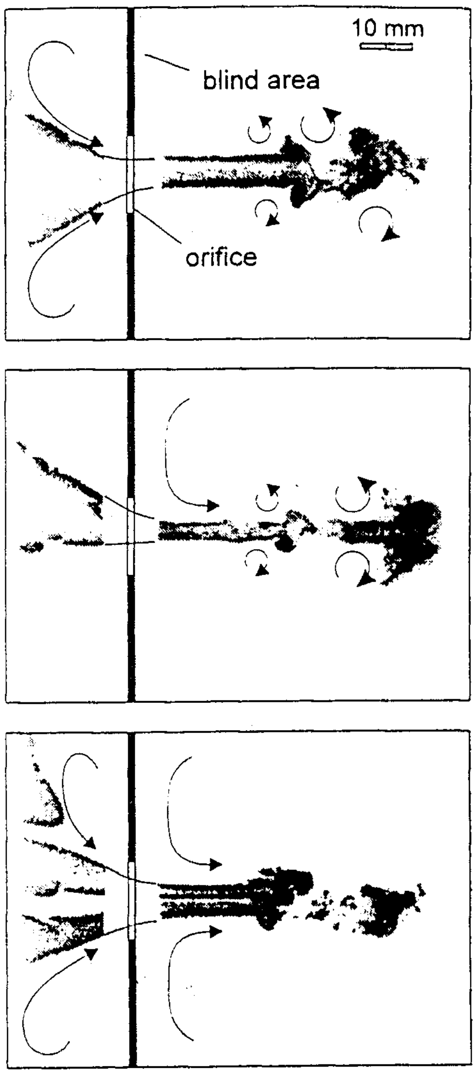

Fig. 11: Igniting phenomena resulting from a jet of burnt hydrogen and radicals in hydrogen-air-mixture

There are several activities in the international nuclear community to develop sophisticated codes for modelling severe accident situations, both in the primary system and in the containment. Examples for this code development are compiled in Table 1. To assess these codes and to improve their physical modelling, well-directed experiments are needed. Within experimental activities, there are still three areas of interest

- mixing and free convection in the containment,
- hydrogen combustion and flame acceleration,
- transport, deposition and resuspension of radioactive aerosols.

Doing such experiments, sophisticated measuring techniques have to be applied, because it is not easy to measure very small gas velocities and to detect accurately aerosol concentration in a gaseous atmosphere and resuspension phenomena.

	ASTEC	
Integrated Codes	VULCAIN etc.	COCOSYS-Models (simplified)
Detailed Mechanistic Codes	**ATHLET/ATHLET-CD**	**COCOSYS**
Special Models	**FRECON** **[FIPREM]**	[RALOC] [FIPLOC] KFK-Codes
	Reactor Cooling System	Containment

Table 1: Codes for Simulation of Severe Accidents

V. FUTURE NEEDS IN NUCLEAR THERMOHYRAULIC RESEARCH

Talking about future needs in nuclear thermohydraulic research, we have to distinguish a little between needs for operating reactors and those for advanced reactors under development. Needs for operating reactors result from questions arising with elaborating more sophisticated codes and with the demand to predict also severe accident scenarios as realistic as possible, i.e. with best estimate assumptions. Recently several national and international organisations are dealing with future needs for nuclear safety research. The Nuclear Energy Agency (NEA) in the Organisation for Economic Co-operation and Development (OECD) just a few months ago published a report on "Nuclear Safety Research in OECD Countries, Capabilities and Facilities" /9/ in which also the needs for thermohydraulic activities are outlined. There are also statements by members of the USNRC staff /10, 11/.

In the OECD report /9/, concern is expressed "about the ability of the OECD member countries to sustain an adequate level of nuclear safety research, even though there was an international consensus on research needs and objectives". Especially "the lack of international support for important new experimental facilities at a time when existing facilities were being decommissioned and their experienced research teams disbanded" is regretted. Indeed there have been several facilities, like BETHSY (France), LSTF (Japan), PKL (Germany), UPTF (Germany) in operation now for a number of years and the question is how long their research program will and can be nationally or internationally financed and whether the know-how, gained by the operating teams, can be preserved. Planning, designing, constructing and operating a new test facility would be an international task and such a test facility could investigate containment behaviour under severe accidents and could prove successful mitigation of consequences of such accidents. Old facilities for studying containment behaviour, like HDR or the Báttelle containment model are out of operation. Codes for modelling physical phenomena, to be expected in the containment during severe accidents, like COCOSYS, need input from experimental data not only for improved physical modelling via separate effect tests, but also for system modelling by the help of integral tests.

Public acceptance of nuclear power in the future will strongly depend on the convincing proof, that even in case of a severe accident with core degradation, radioactivity can be detained in the containment to such an extent, that no serious consequences must be expected for the neighbourhood of the powerstation. This will be a need for operating plants and much more for advanced reactors under design. The containment is the last barrier against radioactive impact to the environment and the barriers before must be guaranteed as well. Therefore also research work for improving our knowledge on thermohydraulic phenomena in the primary system during accidents without and with core disintegration must have high priority. In this connection. In-vessel phenomena will receive most attention.

Special needs for thermohydraulic research activities concerning primary system behaviour will come from the development of existing and new computer codes. Modern computer codes use two fluid formulations, which means, that the conservation laws are formulated separately for each phase and therefore interactions between the phases must be known much more precisely than it has been demanded in the past. Such phase-interface phenomena are

interfacial heat transfer,
interfacial mass transfer,
interfacial friction and momentum transfer and dispersion of the phases.

To do experiments in these fields must be a challenge for universities. To be successful in this research, new measuring techniques have to be developed. In an expert meeting, held at Santa Barbara, March 1997 /12/, very promising new measuring techniques were presented. New probes, working on a capacitive or a conductive basis were developed in France, Japan and U.S.A. to measure the phase-interface area in turbulent two-phase mixtures, especially in bubbly flow. Also new laser-doppler and new hot film probes are used to measure phase boundaries. New absorptions and scattering methods - working with light or radioactive rays - are coming in use again. Last not least ultrasonic devices experienced a very promising development in Japan. Tomographic evaluation of signals from three-dimensionally arranged probes are rounding up the informations. The development of modern measuring techniques must be encouraged and financially promoted. Especially universities should feel responsible for research and development in this field.

The need of better experimental information on interfacial phenomena were clearly pointed out by Kelly /10/, when he demonstrated, that the comparison of RELAP 5 predicted values of the interfacial area concentration with those measured by the McMasters university showed a difference of one order of magnitude. Also predicted and measured values for the interfacial heat transfer coefficient in subcooled boiling were far away from each others, especially for void fractions above 5%. Doing research in the field of interfacial phenomena, therefore is not cosmetics but meets urgent needs.

Decay heat removal under severe accidents, i.e. with temporary failing emergency core cooling systems, has to rely on energy- and mass transport by gravity forces. This results in low flow velocities and phase separation. Single-phase and two-phase natural circulation but also counter-current flow should be studied more in detail. Two-phase natural circulation can be affected by flow instabilities due to the low driving forces and the strong dependency of friction forces on void fraction. Therefore flow stability and heat transfer under unstable flow conditions is another subject of interest. Liquid entrainment and phase separation is not yet well known in buoyancy driven flow and there is still a gap in our knowledge on phase separation in pipe junctions like tees. Thermal stratification is expected under small velocities with emergency cooling water addition, but also in some cases during normal operation of boiling water reactors. This thermal stratification causes additional stress in the nozzles of the pressure vessel. The fluiddynamics of these phenomena are experimentally not assessed.

A wide field of research demand is opened by the design and development for Advanced Light Water Reactors. The LOCA transients of these plants rely upon the passive safety features of the system and gravity driven injection has to provide a stable source for long term cooling. These passive systems drive flows with small heads and a reliable safety analysis consequently requires much more precision than the well known active systems of operating reactors with high pump forces.

Not only new, specially planned experiments are needed to support Advanced Light Water Reactor design, but also codes have to be improved to model the special phenomena to be expected in these plants.

Models for interfacial drag in rod bundles at low pres and low heat flux conditions have to be improved respect to a realistic prediction of the void fraction. In connection one has to have in mind, that the large den differences between water and steam at low pressure ca a special problem in modelling slip ratio between the ses. Under gravity forces and with very low flow veloci already in single phase flow thermal stratification may place and hot water is accumulating in a buffer layer. without convection this hot liquid buffer layer can res from condensation. Codes must be able to handle si density differences to predict the development of the buf layer and the effectiveness of mixing methods. Stratifica on can also occur in the cold leg of the primary syst with natural circulation between core and steam generat In this case, a counter-current flow stratification cou develop. Finally codes must precisely predict the uid/vapour interfaces and also temperature gradie within flowing and stagnant liquids.

Studying very severe accidents, also ex-vessel phe mena deserve further research. Design measures and ac dent management procedures for Advanced Light Wa Reactors are aiming to maintain the integrity of the co tainment as long as possible and to minimise off-site dioactive releases. Priority should be given to experimen work aimed at a better understanding of debris coolabil mechanisms, the spreading of core debris with and withc the presence of water and especially the role of crusts coolability and fuel coolant interaction. With respect hydrogen combustion, there is a need to continue the ev luation of experimental data and their use in validati analytical models. Residual concerns remain in the pe formance of passive recombiners over a wide range environmental conditions. Furthermore, there is still a la of knowledge of hydrogen mixing in the containment a especially in its subcompartments. Analytical methc need improvement and validation.

In spite of the fact, that, stimulated by advanced rea tor projects, many research programs are in progress will be still necessary to keep the current level of effort a sufficiently long time to gain enough and reliable info mation for being able to convincingly guarantee the hi safety level of Advanced Light Water Reactors. The ma difficulties in doing research for severe accident scenari come from the high temperatures, the problems in simu ting sustained heating at these temperature levels, representation of simulant material and the scale limi tions. Increased effort is needed in the modelling, requir to extrapolate with confidence from the small scale of t test facility to the full size of the reactor.

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