NEW DEVELOPMENTS IN NUCLEAR SAFETY PHILOSOPHY AND RESEARCH AND THE ROLE OF LARGE SCALE FACILITIES

F. Mayinger

Technische Universität München, Lehrstuhl A für Thermodynamik,
Postfach 20 24 20, 8000 München 2, Federal Republic of Germany

ABSTRACT

Research for the safe operation of nuclear power plants has always been fitted to the safety requirements. In recent years, the scenarios discussed go far beyond the so-called design basis accident. In risk studies these scenarios were always summarized under the heading of "residual risk". In newer deliberations there is a tendency to develop and provide measures for mastering also accident sequences beyond the design basis accident. The strategy of these measures is called "Accident Management".

Accident management procedures can be derived from the experience gained in experiments performed worldwide. These experiments cover a wide field of operational conditions with respect to incident or accident sequences as well as to the thermo- and fluid dynamic state. For extrapolating the situation to a real plant, however, large scale effects need further investigation.

INTRODUCTION

Nuclear safety research must always be aimed at the safe operation of the plant, therefore, it has to get information from experience gained during operation and during transient or incident situations, e.g. with failing components. Risk studies for nuclear power plants are an especially suitable and proper means to judge the contribution of failed components of the system to the expansion or escalation of incident or accident scenarios. Additionally, consequences for nuclear safety techniques and also for safety research can be drawn from these studies. The collecting of experience here, therefore, is based on systematic combination, physical laws, mathematical probability, and especially also on the operational behavior and reliability of the components integrated in the plant.

Risk studies as far back as the seventies [1,2] have shown possibilities for operational measures and for improving the safety systems, allowing one to master even highly improbable but severe accidents. From risk studies, but also from deliberations to master situations of severe accidents, a new safety concept can be developed which allows one to control accident scenarios (formerly summarized under "residual risks"), and to avoid severe consequences for the environment.

DEVELOPMENT IN NUCLEAR SAFETY STRATEGY

To understand more recent deliberations in nuclear safety better, it may be useful to recall the historical development in the safety strategy for nuclear power plants, especially for water cooled reactors. In the middle of the fifties, the beginning of the peaceful use of nuclear energy in the Federal Republic of Germany, safety deliberations were based on the so-called "Maximum Credible Accident" (MCA). International nuclear safety research then very soon showed that there is a much broader scenario for accident possibilities which, however, can be covered by taking into account a number of comprehensive and enveloping events. These events have been called "design basis accidents" since the middle of the sixties. The protection and safety devices therefore had to be designed to control these design basis accidents [3].

As mentioned before, risk studies promoted the discussion on severe accidents - beyond the design basis accident - in the second half of the seventies. A severe accident is always initiated by an operational transient or by a leak in which, in addition, an independent and comprehensive failure of protection and safety devices is assumed - a failure beyond the extent presupposed in the licensing procedure for minimal availability of safety devices. A severe accident would always develop slowly in water cooled reactors of the western type due to the design principles and due to physical laws. Several hours would pass before an incident or design basis accident would escalate to a severe accident with unallowable radioactive release into the environment. This time can be used for measures to master the situation. These are called emergency measures or "accident management" as discussed and proposed by Herbold and Kersting [4,5] several years ago. A broad spectrum of severe accidents which can be managed and mastered were summarized earlier under the definition of residual risk. It must always be the aim of the accident management to retain the nuclear fuel in the pressure vessel and in the primary system and to keep it in a coolable condition. Former measures against design basis accidents are supplemented by further actions serving the emergency protection and mastering the accident in a comprehensive way.

In this way, the originally deterministic treatment of nuclear safety questions was supplemented by probabilistic considerations in the second half of the seventies, originated by probabilistic risk studies. The licensing procedure, however, is based on a deterministic treatment and probabilistic studies have supplementing character only. This deterministic treatment in the licensing procedure can be followed up in the future by including accident management, and, by this, with mastering scenarios of originally "residual risk" character. In this way, a catastrophic failure of the core can be avoided.

The meaning of accident management is to activate components of operational character for avoiding unallowable thermo- and fluid dynamic conditions in the primary system, or to cool the core in spite of the fact that these components originally were not defined as safety components. Accident management can also mean to reactivate temporarily failed safety components by manual measures.

The safety concept of nuclear reactors has been developed in the past as a multi-level concept according to a "defence in depth" strategy. Recently, a further level has been added to the already existing three level system characterized by accident management procedures, thereby extending the comprehensive and reliable safety measures outlined in Tab. I.

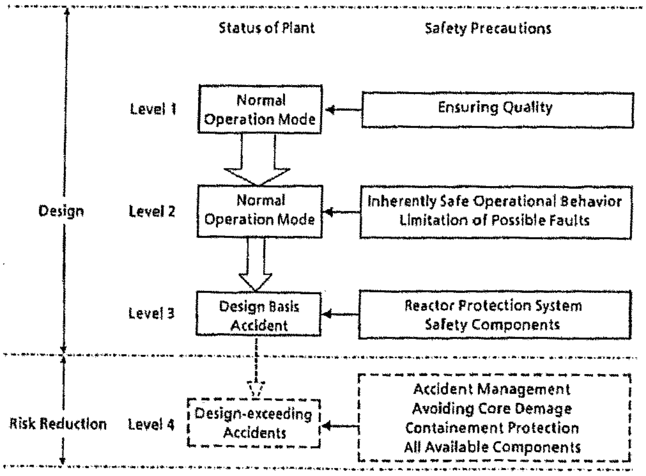
The safety systems of a nuclear reactor are designed and constructed in such a way that they tolerate errors and failures. For example, the effect of human mistakes is identified and indicated by automatic control devices during a certain period. Therefore, they protect the plant against all rationally imaginable accident events. Due to the high redundancy of the safety systems, this function is fulfilled with great reliability, even if an additional failure occurs which is independent from the original cause of the incident. Such an additional failure can be e.g. the breakdown of an emergency cooling pump. In this case the decay heat removal is guaranteed even if an additional pump should be under inspection.

The additional new protection level is formed by the plantinternal accident management. This fourth protection level allows one to control even severe accidents - beyond design basis accidents - within such a limit that no catastrophic core degradation occurs and that full integrity of the containment is assured permanently.

ACCIDENT MANAGEMENT

When discussing scenarios of nuclear accidents, it is useful to introduce a classification of events exceeding the design basis accident which is based on the condition of the core. One may distinguish three categories as given in Tab. II. In the first category of this classification fewer safety systems are available than required in the safety criteria or according to the licensing rules. However, the temperatures still stay within the limits postulated in the safety criteria. In the second category, all those accidents are summarized in which the core is damaged - e.g. some of the pellets are in direct contact with water, or are even partially molten - however, the core as a whole still remains coolable. This category, for example, is representative for the condition of the core in the Three Mile Island Reactor. The third category contains all conditions of a completely molten core having penetrated the reactor pressure vessel and with no possibility of reinstalling any of the cooling systems.

The safety systems for controlling design basis accidents are designed with large margins, and, in addition, the calculations for this design were performed under extremely pessimistic - so-called conservative - assumptions for the efficiency of heat transport out of the core to a



Tab. I: Multilevel concept for safety precautions in nuclear power plants.

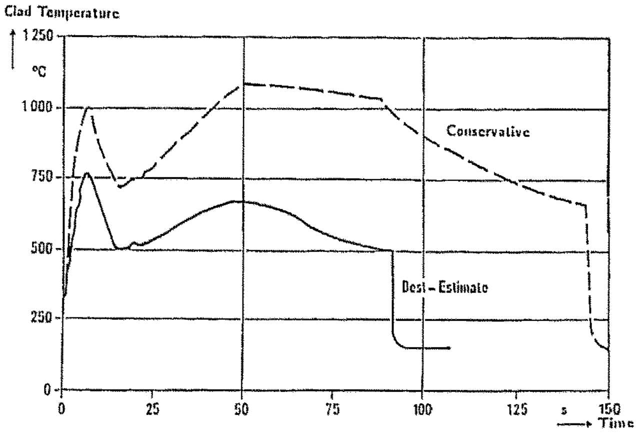
Category I:	Requirements of licensing not fulfilled, but full coolability of the core possible with remaining safety systems. Temperature limits of licensing not exceeded.
Category II:	Accident sequences with severe core damage. By reinforced cooling, however, long-term decay heat removal can be reached (TMI acc.).
Category III:	Classification of accidents (beyond design basis accidents)

Tab. II: Classification of accidents (beyond design basis accidents).

heat sink. The degree of these conservative assumptions can be illustrated by the predicted temperature history of the fuel rod cladding during the blowdown and reflooding period during a loss of coolant accident caused by a double ended pipe break as shown in Fig. 1. The conservative calculation used in the licensing procedure predicts a maximum temperature of the cladding of the fuel rods of nearly 1200 °C. This assumes all accumulators except one which feeds directly to the leakage are available, and if two of four existing decay heat removal pumps feed their water into the pressure vessel. According to this calculation, the cladding of the fuel rods would be rewetted 150 s after the incident started. Nuclear safety research over three decades has provided so much knowledge and information in the meantime, that all states of this incident can be calculated and predicted by a "Best Estimate Analysis" which provides very good agreement with the conditions to be expected under a real accident. According to this analysis, the temperature of the cladding would rise in the very first excursion - at the end of the blowdown period - only up to 750 °C and the temperature curve reaches a second maximum of 650 °C approximately 50 s after beginning of the accident. After 90 s the fuel rods would be rewetted again.

As a consequence of this experience and these considerations, fewer emergency core cooling systems would be needed to guarantee the upper temperature limit of 1200 °C than required in the licensing procedure with its conservative calculations. In Tab. III, the minimum requirements are listed for the availabilty of emergency and decay heat removal systems to assure integrity of the core [6]. For large leaks in the primary cooling system, one could even dispense with the accumulators, which have an extremely high reliability due to their automatic and passive action. One low pressure decay heat removal pump would be sufficient to guarantee the cladding temperatures would remain below 1200 °C [7].

For leaks with smaller areas , the steam generators take over the heat transport out of the core more and more. To allow for this, the secondary side of the steam generators has to be brought to lower temperatures by depressurization. Then on the primary side, the heat transport is provided by natural convection between the core and the steam generator, which starts automatically and without any measures from outside. Depending on a supposed delay of depressurization on the secondary side, in most cases one high pressure injection pump is sufficient to keep the water inventory in the primary system at a required level for a certain period of time. For long-term cooling, an additional low pressure decay heat removal pump is necessary. It should be mentioned here that German pressurized water reactors are provided with four high pressure injection pumps, four low pressure



1300 MW - PWR, 2 A - Break between Pump and Pressure Vessel

Fig. 1: Peak clad temperature after a double-ended break of a primary loop line as a function of time.

Leak Area (cm^2)	H.P. System	Acc.	L.P. Flooding	L.P. System	Avail. Blow Down Tolerance Sec. Side (min)	Feed Water Supply
> 500	-	-	1	1	-	-
200 - 500 300 - 500	1 -	- 2	1 1	1 1	- -	- -
80 - 200	3 or 4 2 1	- - -	2 1 1	2 1 1	- 60 30	- 1 Main Feed
50 - 80	2 1 1	- 3 -	1 1 1	1 1 1	60 60 30	Water Supply or
25 - 50	2 1	- -	1 1	1 1	90 60	2 Emerg.Feed
2 - 25	1 -	- -	1 1	1 1	> 120 30	Water Supply

Tab. III: Minimum requirements for available systems to transport heat out of the core during a LOCA.

decay heat removal pumps, and eight accumulators.

The steam blowing out from the secondary side of the steam generators due to depressurization has to be replaced by injected water. To achieve this, usually three main feedwater pumps, two startup and shutdown pumps, and four emergency feedwater pumps are available. Each of the latter ones is driven by its own diesel engine. Out of all these possibilities for injection feedwater into the steam generators, one main feedwater pump or two emergency feedwater pumps are sufficient to keep the temperature of the core within the requirements postulated for category I (Tab. 2).

If this minimum requirement would not be fulfilled in an emergency case, then the accident would escalate to conditions described as category II. This means the core would heat up, the cladding of the fuel rods would fail, and the core would partially melt. The tolerable time until the melt would start to penetrate the pressure vessel depends on the size of the leak in the primary system, and therefore, on the kind of initiating event. Usually this tolerable time until catastrophic core failure would occur and the melt would start to penetrate the pressure vessel - which would correspond to category III - amounts to several hours. Figure 2 shows this tolerable time for a complete station blackout. This tolerable time would be more than two hours for a pressurized water reactor, during which electric power must be reactivated again by any means. New studies, performed separately for each German power station, have confirmed that the reavailability of electric power within two hours can be assumed with certainty also due to the fact that several plants were provided with additional power feed lines which are fairly independent from the general power grid. In addition, the original capacity of batteries delivering direct current was doubled. These batteries feed all necessary controlling and protecting devices during the accident, including the driving mechanisms for valves needed for depressurization as an example.

The shortest tolerance time for measures against an escalation of the accident would be available for a loss of coolant accident with a large break. Here, however, one has to take into account that due to the primary system, such a break has an extremely low probability, as risk studies have clearly shown.

The tolerable periods until a category II accident would develop into a category III one, can be largely extended by accident management measures [8] as Fig. 3 shows. For a station blackout, accident management prolongs this tolerance time up to six to eight hours, until AC-electric power from outside or from emergency diesel engines must be available again.

The procedure of accident management during an escalating accident shall be demonstrated for the example of a complete failure of all feedwater systems in a pressurized water reactor. Such a situation would occur if all three main feedwater pumps, both startup and shutdown pumps, and also the four emergency feedwater pumps would not be available. This situation would lead to the evaporation of the secondary side coolant inventory of the steam generators, and finally, to a complete breakdown of the heat transport out of the core. The decay heat of the core would then be stored in the primary coolant which would result in a pressure rise until the safety valves of the primary system would open, and a steam-water mixture of the coolant would be blown inside the containment. Losing the primary water, the core would start to become dry and would finally begin to melt.

Engineering analysis and phenomenological simulation of the accident sequence with the help of computer codes show that several installations, originally not considered as safety systems, are available to guarantee that decay heat can be removed from the core, and that the core remains safely in its position in the pressure vessel. All these accident management measures have to start from the strict order that the primary system should remain closed as long as possible. Therefore, at first one has to try to reinstall the feeding of the steam generators. One possibility for achieving this would be to connect the feedwater tank directly with the secondary side of the steam generators. This feedwater tank

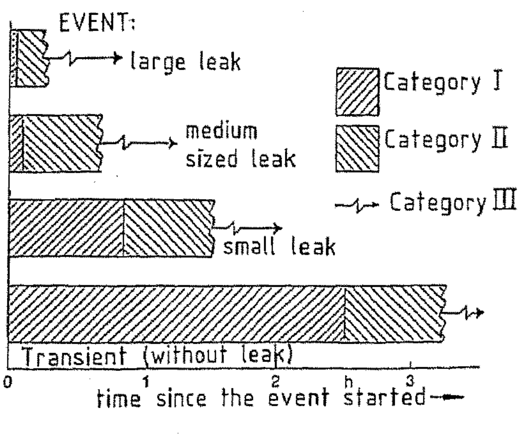


Fig. 2: Tolerance time for accident management measures with various incident sequences.

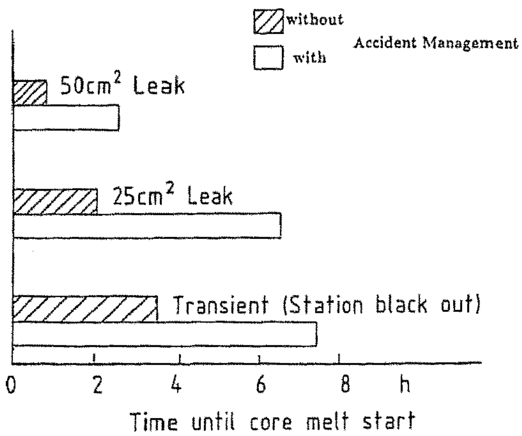


Fig. 3: Tolerance time due to accident management (Measure: Opening of pressure relief valves and by this activating injection of accumulators).

is usually under a pressure of five to ten bar. So, for a while, the water from the feedwater tank could flow to the steam generators without using any pump. Later on, or if no feedwater tank is available, fire extinguishing pumps could be connected to the secondary side system, feeding water into the steam generators from any water reservoir, for example from the fire protecting basin, from a river, or even from the drinking water grid.

Assuming that all these efforts fail, the protection and defense concept of the accident management would not be exhausted yet. One now could start to depressurize the primary system via the relief valves at the pressurizer. This depressurization results in a decrease of the fluid temperature in the primary system via flashing, and by this supports the cooling of the fuel rods. Naturally, the amount of water released in the form of steam has to be replaced again as early as possible, but not later than the time at which the fuel rods start to become dry at the outside of their cladding. One possibility to feed water into the pressure vessel again is to lower the pressure in the primary system below the pressure of the accumulators, i.e. below 27 bar. As a result, the large water reservoir of the accumulators can be used, which guarantees the cooling of the core for at least another three or four hours. Another possibility is to provide the necessary AC electric power supply again for starting at least one of the emergency core cooling pumps.

One could also use this period to bring external pumps to the site which then feed water to the secondary side of the steam generators, starting the natural convection and by this, the heat transport again. The core would be seriously damaged only if all these measures fail within the available time, and the accident would escalate into a scenario of category III.

With boiling water reactors there are also numerous possibilities to avoid core damage in situations which are beyond the design basis accident. Figure 4 shows various possibilities for feeding water into the primary system of a so-called Type 69 German boiling water reactor as an example. With regard to the high pressure injection system, water supply lines for fire protection or for drinking water can be used to reinstall the emergency core cooling. In particular, a pressure injection system driven by a steam turbine (TJ-System) allows one to supply water into the pressure vessel for a very long time. Driven by the primary steam, it is completely independent of the electric power supply, and it works even if all AC power, including that from the emergency diesel engines, is discontinued. The oil for lubricating the bearings of this system is delivered from a pump which is driven by a DC electric motor getting its energy from a battery.

Utilizing these possibilities for feeding water into the primary system, a long period of station blackout, including failure of all emergency diesel engines, can be tolerated. This is illustrated in Fig. 5 where the tolerable and available time for various accident management measures is shown.

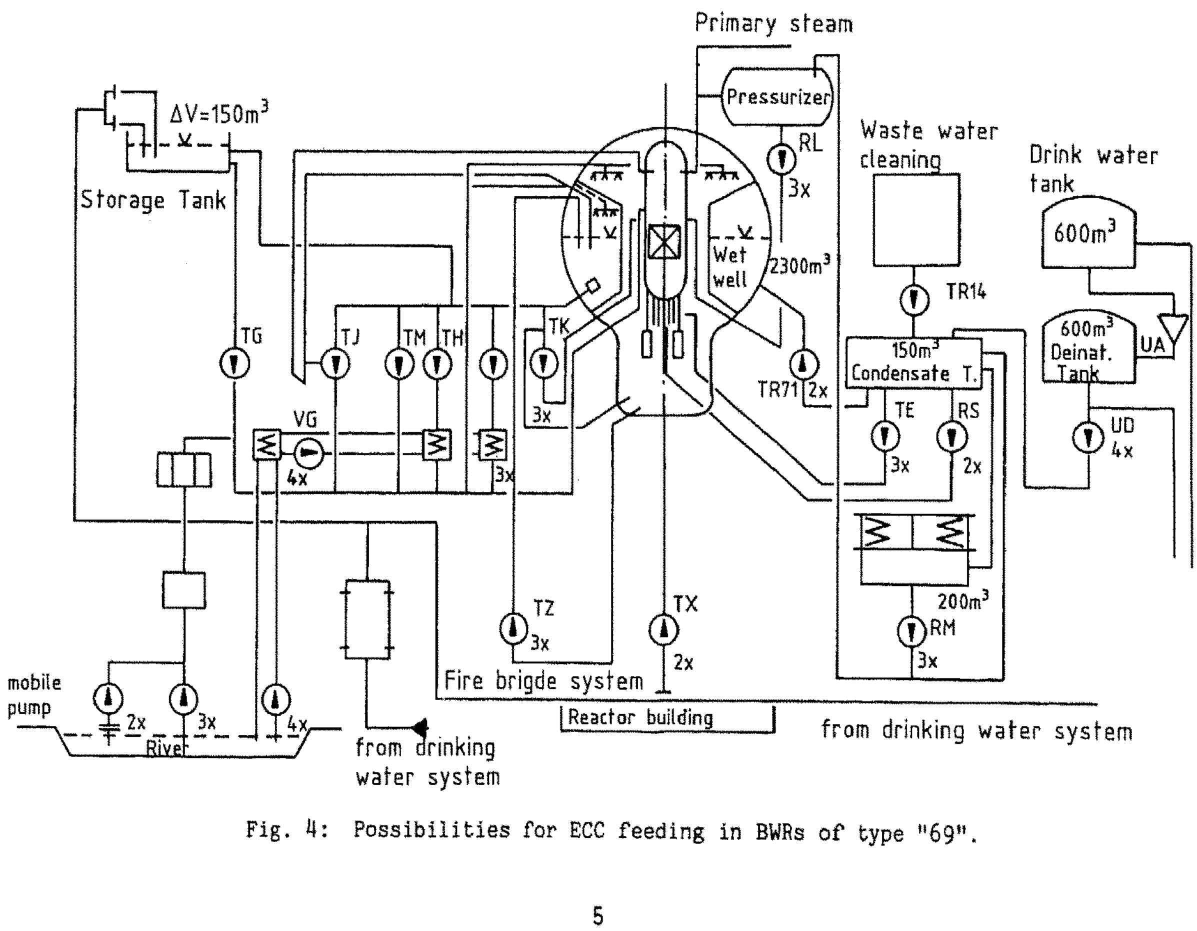


Fig. 4: Possibilities for ECC feeding in BWRs of type "69".

NEW RESEARCH EXPERIENCES FROM LARGE SCALE EXPERIMENTS

Research for the safety of nuclear reactors is and has been performed in the Federal Republic of Germany on a large scale, and according to national and international well balanced plans, since 1972. The research activities were always adjusted to safety requirements. In spite of independent financing and controlling of these safety research activities, there was always guaranteed a close contact with the licensing procedure, and also with the vendors and users of nuclear power plants, to improve a mutual flux of information and knowledge. Important knowledge from the nuclear safety research has found immediate acceptance and application from the authorities and the utilities.

General aims of the nuclear safety research program are

- to widen the knowledge on possible courses and sequences of accidents,
- to develop modells and computer codes for a realistic proof of the safety standards,
- to analyse and demonstrate the safety limits,
- to further develop and optimize nuclear safety techniques.

At present the research activities are concentrated on the following four areas

- components safety and quality assurance,
- operational transients and accident sequences,
- interaction between man and machine,
- risk and reliability.

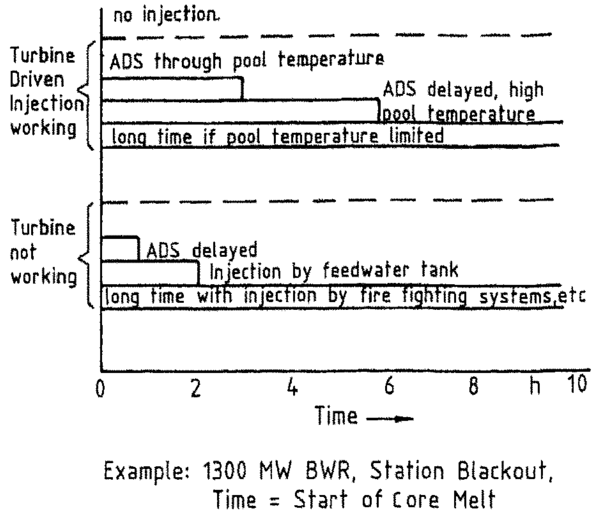


Fig. 5: Effectiveness of accident management measures.

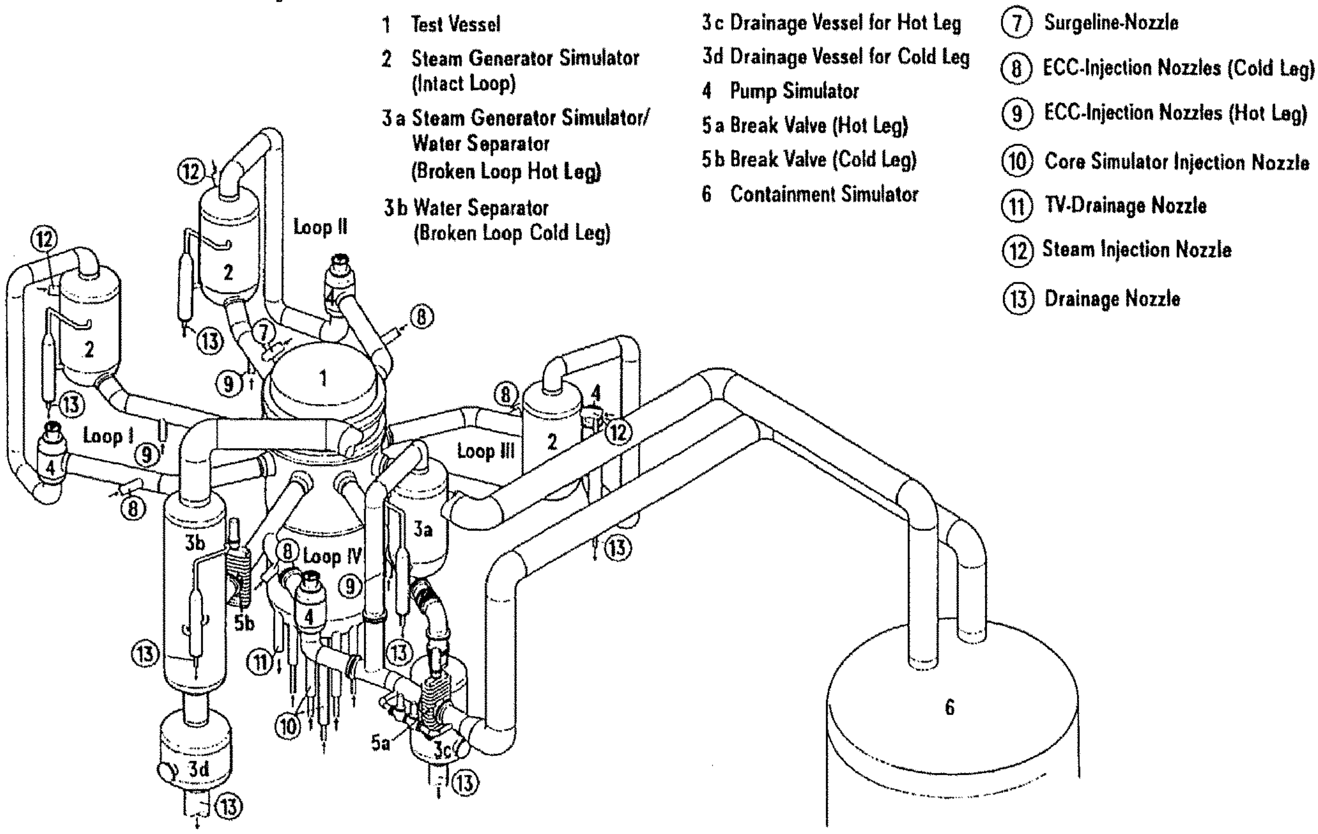


Fig. 6: Upper plenum test facility, primary loop.

It is not possible to discuss all the knowledge and experience gained during more than fifteen years in research activities. Therefore, a certainly subjective and apparently arbitrary example of the research programs shall be presented here. In international cooperation with the US-NRC, the Japanese JAERI, and the BMFT, the so called 2D/3D research program has been performed, aiming mainly at the efficiency of the emergency core cooling system. The German activities within this program have been concentrating on the experiments in the so-called "Upper Plenum Test Facility" (UPTF) [9], shown in Fig. 6. In this original scale test facility of a 1300 MW Pressurized Water Reactor, the fluid dynamic phenomena of the emergency core cooling process in the primary loop are being studied. Fields of special interest are the upper plenum and the down comer in the pressure vessel, and the primary piping system including the in- and outlet part of the steam generators. In this program, accidents of a wide variety, from small leaks up to the double ended break, are investigated.

In case of a small leak, or in case of an incident without leak but with a station blackout, the natural convection in the primary system between the core and steam separators plays the dominant role for the energy transport out of the core. Under certain circumstances, fluid dynamic situations are conceivable in which the fluid is not circulating via the hot leg, the steam generator, the pump and the cold leg, but in which only the hot leg, i.e. the primary pipe between the pressure vessel and the steam generator, is available for the natural convection and for the heat transport. Under such conditions, the steam produced in the core flows through the upper part of the hot leg pipe to the steam generator, condenses in the rising part of the U-tubes, and then the condensate flows back along the bottom of the hot leg pipe. This special convective situtation is called reflux-boiler-condensor mode. This mode of heat transport would be disrupted - and by this also the heat transport out of the core - if the rising steam would prevent the condensate from flowing back into the pressure vessel by shear or momentum forces. Experiments in small scale facilities have shown such situations of a Counter Current Flow Limitation (CCFL) could occur in the hot leg around the mass flow rates which would be expected or have been calculated by computer codes for the reflux-condensor mode.

Experiments in the UPTF-facility have demonstrated that there is a strong scaling effect on the CCFL. Figure 7 shows this scaling effect. In scaled down facilities, CCFL conditions have been observed at 16.5 kg/sm^2 and 26 kg/sm^2 vapour mass flux, while the reflux-condensor mode could be maintained in the UPTF facilities up to 36 kg/sm^2. These typical mass flow rates of the reflux-condensor mode, also to be expected in a reactor, are much lower than the UPTF value, which means that in case of a station blackout or a small leak, this heat transport mechanism between the core and the steam generator would work safely. Figure 8 gives the same results in the so-called Wallis diagram where the superficial steam velocity is plotted versus the superficial water velocity. Also in Fig. 8, data used in the

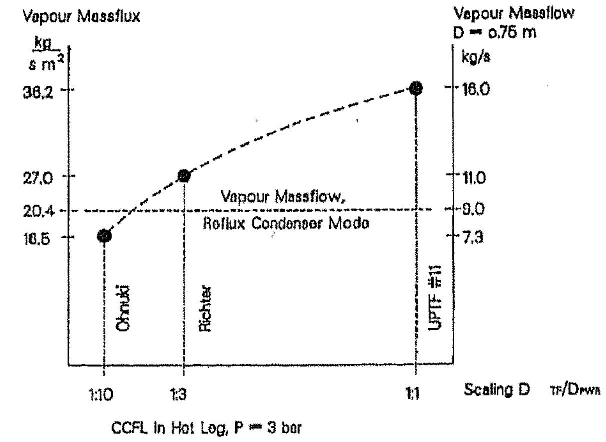


Fig. 7: Scaling effect on countercurrent flow limitation during reflux condenser mode.

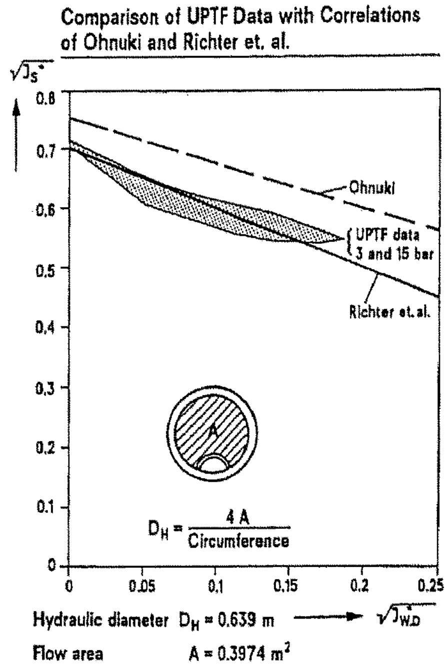


Fig. 8: Countercurrent flow of steam and saturated water in the hot leg (UPTF) [10,11].

licensing calculations for the so-called Konvoi Reactors is given. The licensing assumptions for Counter Current Flow Limitation are at much lower steam velocities than the experiments in the UPTF facility under realistic conditions and at fully scaled size.

The phenomena in the upper plenum can be tested in a realistic situation only in the UPTF facility. For reactors with only cold leg ECC water injection, the transport of the liquid carried over into the upper plenum is of interest. For reactors with ECC hot leg injection, it is of great importance to know how much of the steam is condensed by the cold ECC water, so as to avoid steam binding. For both cases, the water falling back to the core through the tie plate supports the core cooling considerably. The fluid dynamic situation in the upper plenum with hot leg injection is roughly sketched in Fig. 9. The steam produced in the core and flowing up to the tie plate tends to escape through the broken hot leg, and the ECC water accumulates on the tie plate mainly near the location of its injection. Preferably it will also break through to the core there.

The situation of downward water penetration and upward steam flow at the tie plate with hot leg injection from three loops and the fourth hot leg, assumed to be broken, is shown in Fig. 10. The cold water coming from the ECCS can penetrate the tie plate, and in this way reach the core at distinct locations, whereas the steam escapes upward between these areas. This means that there is really no steam binding in the upper plenum, and the ECC water can act very early as a coolant for the core. The situation becomes even clearer if one looks to the distribution of the fluid temperatures at the tie plate level, as shown in Fig. 11. The penetrating water is highly subcooled and has, therefore, still a great capacity for condensing steam in the core. This steam condensation in the core or at the core level promotes a strong cross mixing which improves the core cooling.

The counter current flow conditions on the cold leg side and in the down comer during emergency core cooling are shown in the Figs. 12 and 13, as observed in the UPTF facility. At low ECC water mass flow rates, stratified flow exists in the cold leg, and the ECC water smoothly flows to the down comer of the pressure vessel. Due to the simple injection mode in the cold leg, however, with high water mass flow rates, the ECC water can form a plug in the cold leg which blocks the steam in the down comer. As a result, much of the ECC water is transported to the circulating pump, not reaching the pressure vessel and the core. Both situations can occur at the same steam mass flow rate. They are mainly, or even solely, caused by the water mass flow rate.

In the down comer of the pressure vessel, the steam produced in the core and tending to flow up the down comer, may prevent the ECC water from reaching the core. The steam plug moves the ECC water to the broken loop. This phenomenon is sketched in Fig. 13. This process may impair the efficiency of the ECC system. The overall system behavior during the end of the blowdown phase, refilling and reflooding of the core with a double ended break in the hot leg, is given in Fig. 14. The experimental procedure in the UPTF facility needs a short conditioning phase before the situation corresponding to the blowdown phase in the reactor is reached. In Fig. 14 the period of this conditioning phase is 37.5 s.

The figure shows also that in case of such an accident 7 s after the end of the blowdown phase, the core is flooded to a great extent, and that 30 s after the blowdown phase the core would be quenched already. The first water reaches the core via the hot leg injection durig the blowdown phase. This means that the cooling process starts much earlier than predicted in the licensing

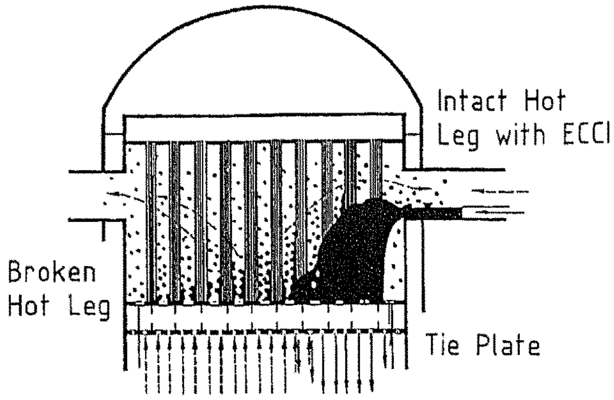
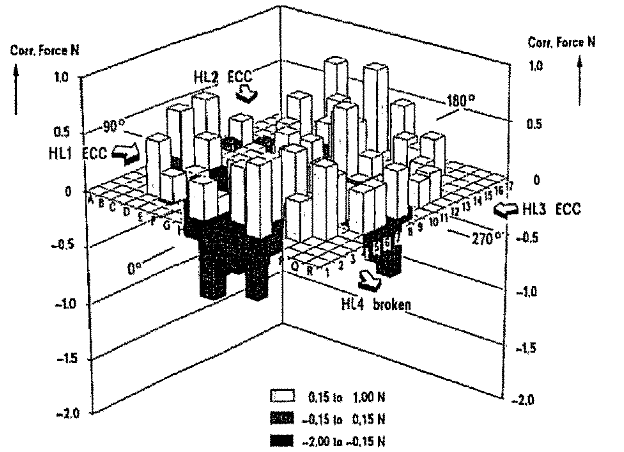
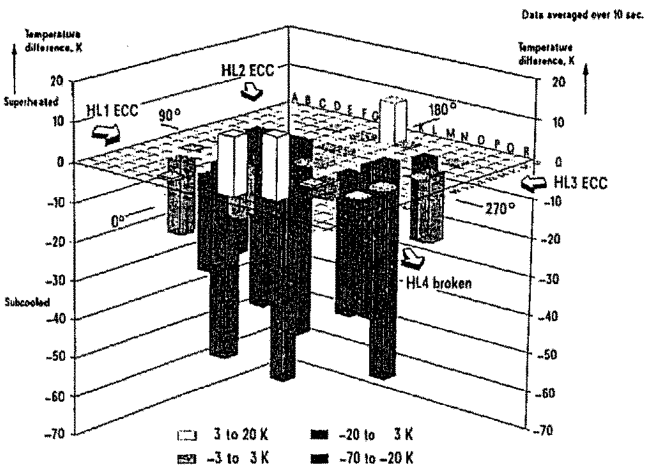


Fig. 9: Phenomena in the upper plenum with hot leg ECC-injection.



UPTF Test No. 12 (RUN 014) – Discrete Values of Corrected BTD Forces at 200 s
(Set Tieplate JKF = 0 JNG = 1200 QHA 05 = 45 kg/s)

Fig. 10: Water penetration through the tie plate with hot leg injection (given in momentum force on break through detectors) (UPTF).



UPTF Test No. 12 (RUN 014) – Discrete Values of Fluid Temperatures
10 mm below Tie Plate at 200 s

Fig. 11: Subcooling of water penetrating the tie plate downwards with hot leg injection (UPTF).

calculations.

Small scale facilities have shown a much larger holdup of liquid in the upper plenum and, by this, a later and lower efficiency of ECC water than would be the case in a real plant. This was one of the main and important results obtained form the experiments in the UPTF.

CONSEQUENCES FOR FUTURE NUCLEAR SAFETY RESEARCH

Accident management actions need clear and reliable information about the situation in the primary system of the reactor. Such information includes, for example, the water level in the pressure vessel, the pressure and its tendency - rising or falling -, the temperature above the core, and also the pressure in the containment. The most important information about the cooling situation of the core is the water inventory in the pressure vessel. Therefore, all German pressurized water reactors are equipped with liquid level detectors covering the region from the upper fuel element end boxes throughout the whole dome of the pressure vessel. These liquid level detectors work according to the principle of resistance thermometers.

In addition to this information, however, the efficiency and the feedback of accident management procedures must also be known in advance. Therefore, a research program studying transient and accident management situations is presently under discussion in the Federal Republic of Germany. If this program becomes active, it will be performed in the UPTF facility mentioned above.

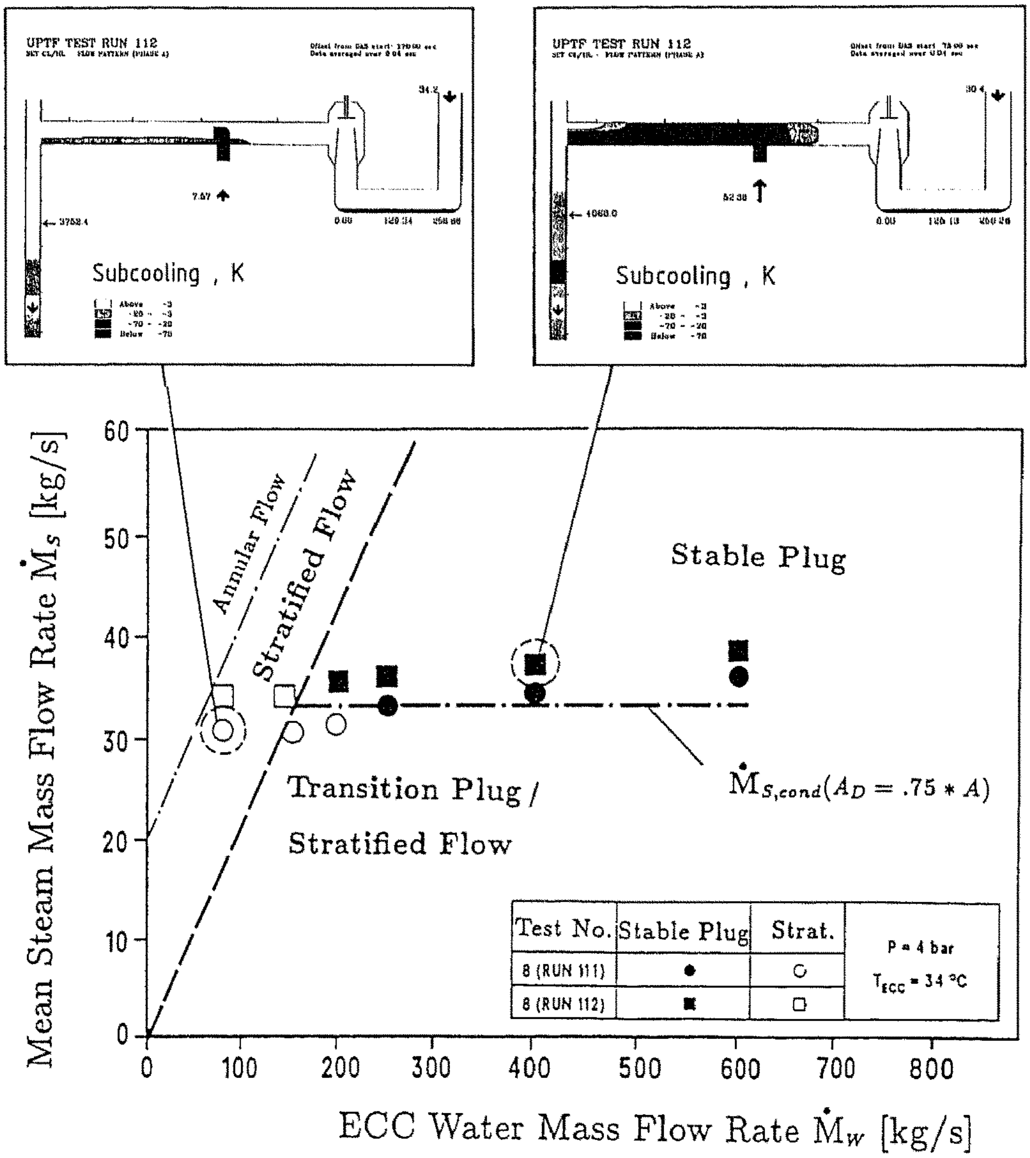


Fig. 12: Flow pattern in cold leg during ECC-injection (UPTF).

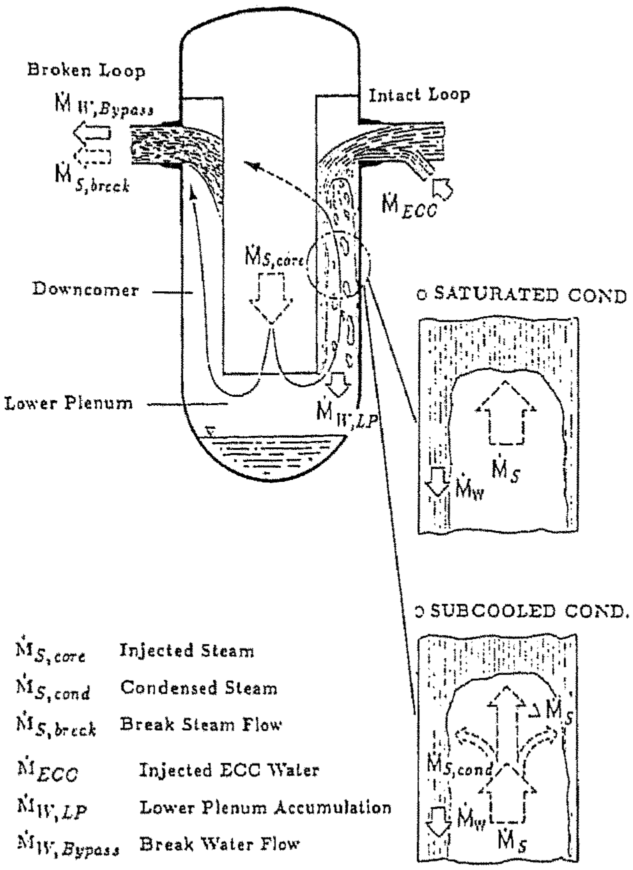


Fig. 13: Flow phenomena in the downcomer under countercurrent flow conditions (UPTF).

Phenomena which could be studied in such a program are for example

- water penetration into the core with hot leg injection,
- mixing of the cold water injected through the hot leg with the water inventory in the primary system,
- formation of separated flow in the hot leg,
- flow through and to a small leak in the hot leg and the related energy transport,
- pump loop seal clearance,
- Reflux-condensor mode by means of injecting water through the high pressure emergency core cooling system.

Experiments studying the integral behavior of the plant during accident management procedures could be

- study of the flow conditions in the primary system with loss of all secondary side heat sinks, especially study of the mass and energy transport from the core to the pressurizer and blowdown behavior of the relief and safety valves,
- study of thermo- and fluid dynamic conditions with depressurization of the primary system and feeding of ECC water via the high pressure injection pumps,

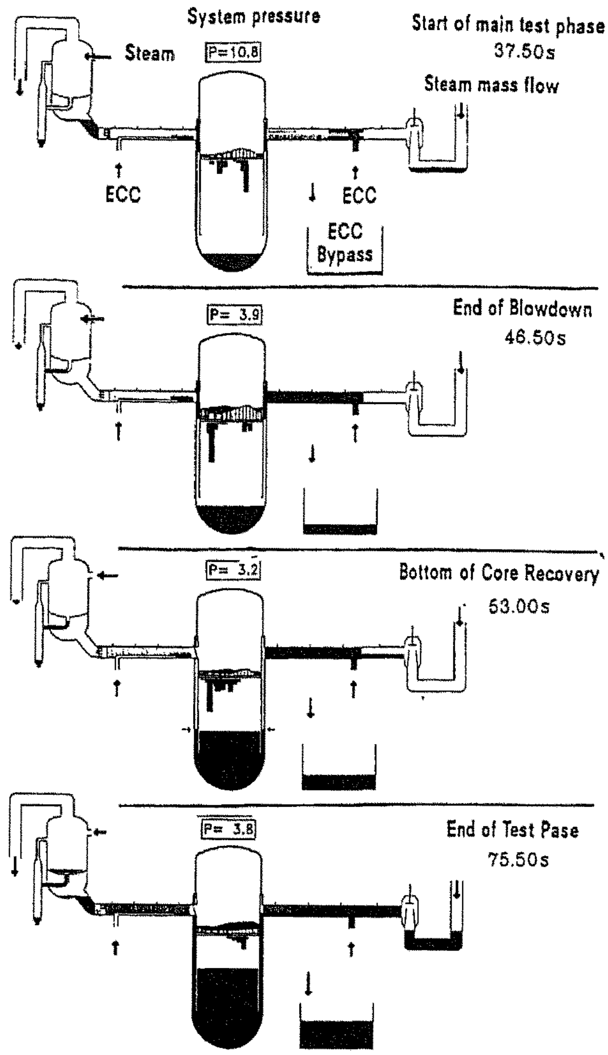


Fig. 14: Overall system behaviour during refilling and reflooding with combined (cold leg and hot leg) injection (UPTF).

- study of the periodic injection of water from the accumulators activated by the depressurization of the primary system below 27 bar.

Up to now, there has been no final decision about this research program. According to the opinion of the author, it could be a very valuable counterpart to the experiments already planned or carried out in the French BETHSY-facility, or in the Japanese ROSA IV plant. These two facilities can work with the full pressure to be expected in a pressurized water reactor during a severe accident, however, they are scaled down by a factor of approximately 100. The UPTF facility is designed for low pressure - maximum pressure approximately 20 bar -, however, it has the full size of a 1300 MW pressurized water reactor. The density ratio between water and steam at situations to be expected during severe core accidents varies between 1/300 and 1/15. The density ratio in the UPTF facility could be varied between 1/300 and 1/80. Experience in nuclear safety research has shown that there is more basis for extrapolating experimental data via pressure and density ratios - especially to higher pressures - than to scale from measured results obtained in small facilities to the full size of the reactor. Existing data taken in high pressure facilities and also experiments planned in such facilities could be used for pressure extrapolation. For scaling to the full size, however, it must be taken into account that flow characteristics can change completely from a small scale pipe or vessel to the full size of the plant. Therefore, geometrical scaling may sometimes be risky, however, from the licensing point of view, experiments in small scale facilities usually give conservative results. But for considerations with respect to accident management, conservative information is of restricted value because actions during an accident management procedure can only be optimally planned if the real situation and the real thermo- and fluid dynamic behavior in the plant is known. Therefore, experiments in a facility of 1:1 scale are of high priority.

Within the frame of large scale experiments, the research program in the HDR facility has to be mentioned also. This HDR progam will focus its future phase on the thermohydraulics during blowdown, the dynamic situations during earthquakes, the dynamic situations during shock pulses from outside, and also with the problem of material stresses due to thermal shocks.

It must be mentioned also that basic research is becoming a wide field of activities with the analy is of thermo- and fluid dynamic problems connected with the study of severe accidents, and of the efficiency of accident management procedures. It will be even more important for these studies than it was in the past to know the real physical behavior, and to describe the physical processes in a realistic and reliable way. These mathematical descriptions must have a good basis for extrapolating to situations beyond the measured region.

It is not only desirable, but necessary, to develop fast running codes describing and predicting the thermo- and fluid dynamic situation in the primary system during severe accidents, with and without accident management procedures in real time. The existing thermohydraulic codes for loss of coolant accidents are too slow by at least one order of magnitude. They also cannot handle situations with severe core damage. Fast, reliable, and physically realistic results from such codes requires the development of better physical models describing separate effects on heat transfer and fluid flow.

Thus, nuclear safety research is a task to be continuously and systematically planned. It has to cover sporadic events a priori, and is not to be influenced emotionally by unexpected transients or incidents. As is generally true in the technical world, research in nuclear safety is a continuous challange, not only to preserve the technical standard, but also to improve it according to the requirements at present and in the future.

REFERENCES

[1] N.C. Rasmussen, "Reactor Safety Study - An Assessment of Accident Risks in US-Commercial Nuclear Power Plants", USNRC WASH 1400 (NUREG-75/014).

[2] Deutsche Risikostudie Kernkraftwerke, Herausgeber: Der Bundesminister für Forschung und Technologie, Verlag TÜV Rheinland, Köln, 1979.

[3] F. Mayinger, Ansprache des RSK-Vorsitzenden, Festkolloquium aus Anlaß des 25jährigen Bestehens der RSK, Bericht GRS-55, Köln, 1983.

[4] G. Herbold, E.J. Kersting, "Analysis of a total loss of AC-Power in a German PWR" (Proc. 5th Int. Meeting in Thermal Nuclear Reactor Safety Karlsruhe, 1984), Rep. KfK-3880/1, Vol. 1, Nuclear Research Center Karlsruhe, FRG, S. 436-447, 1984.

[5] G. Herbold, E.J. Kersting, "Analysis of an Operator Initiated Primary Depressurization in Case of Total Loss of Feedwater Supply" (Offprint Specialist's Meeting on Small Break LOCA Analysis in LWR's Pisa, 1985), (Giardini Editori e Stampatori, Ed.) Pisa, Italy, 1985.

[6] F.W. Heuser, H. Hörtner, E.J. Kersting, "Risikountersuchungen zur Sicherheitsbeurteilung von Kernkraftwerken", in: Sicherheit und Unfallbeherrschung bei DWR- und SWR-Kernkraftwerken, INFORUM, Bonn, Dez. 1987.

[7] K. Orth, "Das Sicherheitskonzept deutscher Kernkraftwerke", in: Sicherheit und Unfallbeherrschung bei DWR- und SWR-Kernkraftwerken, INFORUM, Bonn, Dez. 1987.

[8] A. Birkhofer, "Anlageninterner Notfallschutz in Leichtwasserreaktoren", 4. Jahrestagung der VDI Gesellschaft Energietechnik, Fachtagung IV, Schadensverhütung in energietechnischen Anlagen, Hannover, 24./25.2.88, VDI-Verlag, Düsseldorf, 1988.

[9] P. Weiß, "UPTF-Experiment, Anlage und Ergebnisse - eine Zusammenfassung", UPTF-Fachtagung, Mannheim, 9.2.88, Bericht U9316, Siemens/KWU, 1988.

[10] A. Ohnuki, "Experimental Study of Countercurrent Two-Phase Flow in Horizontal Tube Connected to Inclined Riser", Journal of Nuclear Science and Technology, Vol. 23, No. 3, pp. 219-232, 1986.

[11] H.J. Richter, G.B. Wallis, K.H. Carter, S.L. Murphy, "Deentrainment and Countercurrent Air-Water Flow in a Model PWR", Hot-Leg, NRC-0193-9, 1978.