SEVERE ACCIDENT ISSUES, REGULATORY IMPLICATIONS AND STATUS OF RESEARCH AND DEVELOPMENT IN THE FRG

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1. The basis of licencing in the FRG

Basis for licencing in the Federal Republic of Germany is the "Atomgesetz" (atomic law). There is a careful distinction between damage and danger in this law. In § 1 it is stated that "life, health and neuter goods shall be protected against dangers from nuclear energy". However, dealing with the design and the construction of a nuclear power plant, the law is not using the comprehension of "danger" but is oriented on the comprehension of the "damage". It states that precaution is to be taken against damage and these precautions have to be based on the status of science and technology. In the law there is not stated which residual risk is allowable or acceptable.

The highest court of the Federal Republic of Germany emphasizes in its conclusion concerning the construction of the German fast breeder SNR-300 that the atomic law is not demanding an absolute safety and states that an inevitable residual risk has to be carried by everybody as a social adequate burden. The atomic law gives no statement which residual risk may or can be accepted in the licencing procedure.

So it is within the responsibility of the executive authorities to define the residual risk which should be accepted by the public. Licencing institutions are the authorities of the country in which the nuclear power plant is contructed, and the Federal Minister of Inner Affairs.

It was and it is practice in the Federal Republic of Germany to distinguish between

1. danger - precautions against danger and
2. residual risk and minimization of this residual risk.

So two areas are distinguished and the second area - the minimization of residual risks - is mainly within the judgement of the authority.

Technical details for licencing are given in safety guidelines and safety criteria elaborated and edited by the Reactor Safety Commission and by the Federal Minister of Inner Affairs. These guidelines and criteria deal mainly with precautions against dangers from nuclear plants and give licencing rules

1. to guarantee a safe and environmentally compatible operation,
2. for measures to govern incidents in their earliest stage of development and
3. for measures to mitigate the consequences of an incident.

This, however, does not mean that the German Reactor Safety Commission and also the authorities of the Federal Republic and of the countries are not concerned with consequences of hypothetical severe accidents like core melt down. Risk studies and reliability analysis were discussed in the German Reactor Safety Commission long before the TMI-accident occurred. Important consequences of these risk deliberations were for example the installation of an automatic controlling device for blowing down the secondary side of the steam generators in case of a small leak, and improvements in the valve behaviour.

2. Deterministic or probabilistic procedures in licencing

In the early history of nuclear power plants the low level of experience suggested a deterministic procedure with licencing these plants. The safety of nuclear installations was proved and guaranteed by fixing maximum events of incidents which have to be mastered by the safety devices. This is a proceeding which was and is international practice.

With reference to the increased experience in operating nuclear power plants today this deterministic procedure is often put in question from a probabilistic point of view. Especially questions are rised with respect to sequences of events which go beyond the defined limits of loads and which may lead to serious consequences to the environment of the plant. Risk studies deepened these questions and treated implicitly the problem of safety goals.

Undoubtless there is the possibility today to describe conceivable but hypothetical accident sequences physically meaningful, based on the existing experience with the operation of nuclear power stations and their components, as well as with the available experimental research results and the theoretical knowledge. There is, however existing the problem of completeness of the problem definition with risk analysis. Even if the problem is well defined, still the question is open whether all important events and data are correctly taken in account in connection with the probability of the event to be expected. Finally, each risk analysis is affected with subjective elements - at which subjective has not to be interpreted here as arbitrary -, but the subjectivity is implemented by the fact that the experience of the person evaluating the data and the events plays a certain role.

Therefore, a lot of arguments are existing to stay with the proved deterministic principles for the layout of the plant and the judgement of the safety, in connection with licencing nuclear power stations. Essential reasons are the accuracy and reliability of the safety judgement, and the effectiveness of proving. Besides this it is imaginable to work out a concept for the definition and the use of probabilistic safety criteria, which may be later on introduced into the safety guidelines and which, finally, help to supplement the deterministic criteria.
Knowledge gained from probabilistic studies, however, is a very valuable and important help to find weak points in the plant, and by this to improve the reliability and also the safety. From the beginning it was the working principle of the German Reactor Safety Commission to exclude accidents a priori, thereby, that by improving the reliability the probability of incidents due to weak points in the plant is reduced and that in case of an incident the operating controlling system brings back the behaviour of the plant to normal operating conditions. Beyond this the system of nuclear safeguards is acting as a safe barrier. The German Reactor Safety Commission follows up since a long time the principle of "basic safety" and especially in the last few years essential progress was made in cooperation with the venders. Reliable lay-out calculations based on the latest status of knowledge together with a material selection, especially adapted to the nuclear demand, a careful fabrication accompanied by permanent material testing, as well as periodically repeating testing during the operation period, guarantee a mechanical status of the pressurized components of the reactor which allows to exclude large breaks and helps to narrow the spectrum of incidents considerably.

Parallel to the improvement of the basic safety, deliberations and measures were taken to enforce the instrumentation and the diagnostics for incidents. This principle, followed since years by the German Reactor Safety Commission, may have made some contribution to the fact that German pressurized water reactors are within the leading group of the world list with respect to availability. An important contribution to the high level of reliability and safety certainly also the risk studies did make. The German risk study, phase A, for example showed 40 possibilities to improve the system, which are realized almost completely in German pressurized water reactors in the meantime.

It is well known that the man-machine-relation plays an important role for the safe operation of a nuclear power plant. To improve the reliable behaviour in case of and against actions of the operator, a high grade of automatization is realized in our reactors. Automatic safety systems are co trolling the plant even under incident co ditions, at least for the first 30 minut before an action of the operator is nece ary.
In the future two areas of subjects may p an important role in the discussions of t Reactor Safety Commission. One area of hi interest is the reflection from results o the risk analysis on to the recommendatio and formulations in the safety guidelines and another one is the question of possib simplifications in the licencing procedu which could be answered from the operati experience of existing light water-coole reactors.

A probabilistic judgement of the nuclear safety, therefore, may be concentrated a in some respect also limited to question how well balanced a safety concept is, a how large safety reserves it contains. T risk orientated methods analyzing the ac dent sequences and the reliability, ther fore, offer very important means to iden possible protection measures for mitigat the accident consequences. A very import contribution is expected from the risk s phase B, which is just under way.

3. Core failure accident pathways, status of knowledge

As well as from the US risk study /1/ as from the German one /2/ it is well know that small leaks in the primary system a the loss of outside electrical power con tribute much more to the risk of a nucle power plant than the loss of coolant acc dent due to failure of a primary pipelin i.e. a large break. However, also small do not have necessary and "per se" the c sequence of a core failure; furthermore, additionally another system function wou have to work in a not proper way or to s a wrong reaction. In case of a small lea the reliable action of the pressure reli valves and of the feed water system is e tremely important, as table 1 shows /2/.

Transient	Contribution to core melt probability	Percentage of feed water system and sec. side blow-down	Contributio of feed wat system to p of core mel
small leak in prim. loop	67%	90%	60,3%
loss of el.power	15%	100%	15 %
loss of feed water system	3%	100%	3 %
small leak in press.with loss of el. power	7%	20%	1,4%
small leak in press. with other transients	2%	25%	0,5%
ATWS-incidents	1%	5%	0,05%
sum	95%		80,25

Tab.1: Contribution to risk by small le and transients without loss of coolant

The depressurization of the secondary side of the steam generators, however, must not go on too rapidly because in this case the safeguards would interpret this intended depressurization as a break in the steam-line which would activate the insulation valves. This again would make it impossible to decrease the temperature on the secondary side of the steam generator at least for a while. On the other side the depressurization must not be too slow, because then the energy transport from the core to the steam generator may not be good enough. This was the reason for installing the above mentioned automatic depressurizing controlling device.

Results of risk studies, however, may be also misleading in case of using them for defining proper actions against hypothetical accidents. Both risk studies /1,2/ proceed from the assumption that core melt down is always then the consequence, if the criteria of the safety guidelines are not fulfilled, i.e. if the temperature of the cladding in the core exceeds 1200°C, or if more than 2 emergency core cooling systems fail.
Looking to severe nuclear accidents we first have to keep in mind that licencing calculations start a priori from conservative assumptions and are often far away from physically realistic behaviour. It is well known that the lay-out incident - double ended break - is calculated with a number of conservative assumptions like 20% too high decay-heat, too high peaking factors and too low heat transfer between fuel and coolant.
With large breaks the action of the accumulators plays the most important role in the first minutes after the incident for cooling the core and by this for avoiding unallowable temperatures of the cladding. These accumulators are completely passive systems, which means that they have a very high reliability. German pressurized water reactors of newer design have 4 accumulators.
With decreasing number of effective accumulators the time, until the fuel-rods in the areas of highest power are rewetted, is increasing. However, even under the very pessimistic and physically certainly unrealistic assumption that only one of the 8 accumulators becomes effective and only one of 4 low pressure decay heat removal pumps is working, the rods even in the hot channel area are rewetted after 6 minutes. During this time no temperatures in the core are to be expected which go beyond an unallowable condition. Best estimate calculations performed in extensive studies by the GRS /3/, KWU /4/ and BBR /5/ showed for a 2-F-break a maximum temperature between 1220°C and 1260°C, if only one low pressure pump is effective. Under these thermodynamic conditions certainly a part of the cladding is ballooned and burst, however, all fuel rods remain coolable and no core melt occurs.

One can now assume that this only one and last low pressure heat removal pump is not working immediately but delayed, which would be the case if the outside electrical power fails and all 4 emergency Diesels would not start immediately. Also such delays would be tolerable without uncontrollable core melt, as table 2 shows.

LARGE BREAK				
break size	availability of systems			maximal cladding temperature
	high pressure inject. pump	accumulator	low pressure heat rem. pump	
2F. cold leg	-	-	1	1250°C
2F. cold leg	-	7	-	after 32 min 1900°C
				after 29 min 1200°C

Tab.2: Tolerable activation delay of the low pressure heat removal pump until local core melt would occur

If one can assume that 7 accumulators out of 8 are effective, 30 minutes of time are available until one of the low pressure heat removal pumps is needed. Only after this time the temperature of the cladding in the hot channel areas is rising beyond the limit of 1200°C postulated in the licencing rules, and not earlier than after 32 min local melting areas in the core could occur. The reason for this long time of tolerance is to be found in the fact that the water inventory brought into the pressure vessel by the accumulators has to be evaporated before a temperature rise of not allowable extent can occur in the core.

3.1 Small leaks

Discussions in connection with the TMI-incident and its consequences with respect to the core failure caused the impression in the public, as if such incidents and their sequences - i.e. small leaks in the primary system - were not or not enough taken into account in the licencing procedure and in the safety analysis of the emergency core cooling. In the Federal Republic of Germany years before the TMI-incident occurred, sequences of small leaks were treated intensively and also calculations were performed from which very soon clearly came out that the heat removal out of the core has to be mainly performed by free convection via the steam generators. This is done as mentioned by depressurization of the secondary side of the steam generator. Within this procedure it is not at all necessary that the primary circuit is filled completely with water; furthermore, a vapour-liquid-mixture can be allowed to be formed, without impairing the cooling in the core.

Even if the water level would fall below the upper end of the core, the fuel rods would be coolable, as experiments in a German test loop - the so-called PKL-test-rig - showed /6/. The fluid dynamic conditions in the core, measured in the PKL-test-rig with only cold-side emergency cooling water injection, are presented in figure 1, representing an 80 cm² leakage in the area between primary cooling pump and pressure vessel of a 1300-MW-reactor /6/.

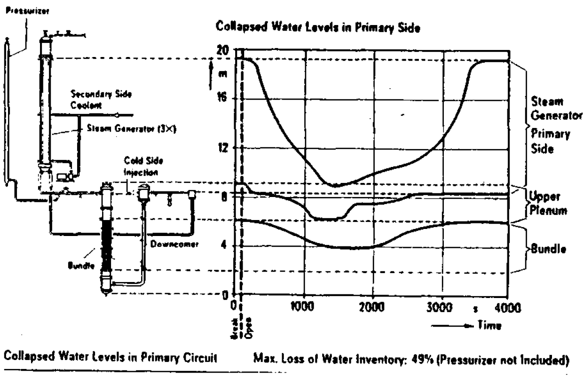

Fig.1: Water level in primary system according to PKL test, small leak *(= 80 cm²) cold leg injection*

In spite of the fact that the primary side of the steam generator, as shown in Fig.1, is completely empty at least for several minutes, a heat transport from the core to the steam generator takes place and the core is cooled. The so-called collapsed water level in Fig.1 falls in the reactor pressure vessel below the mid line of the core without damage to the fuel rods. Here one has to be aware of the fact that the swell level due to bubbles in the liquid is still above the top of the core. This means that the fuel rods are still wetted. *This is a fluid dynamic situation of normal operating conditions in all conventional* - i.e. coal or oil fired - boilers. The liquid inventory in the upper plenum of the pressure vessel is accomplished by the entrainment, due to boiling phenomena in the core which carry droplets through the tieplate which are then deposited.

More favorable conditions with respect to the swell level in the primary system occur if the emergency core cooling water is injected in the upper plenum directly instead via the downcomer with cold leg injection. As well in the steam generator as in the reactor pressure vessel the swell level is falling much less compared to cold side injection, as shown in figure 2. This can be physically explained very simple. The water is now arriving first in the core before it can be carried out via the leakage again, whereas before, a great part of the emergency core cooling water flew immediately to the leak before becoming active as coolant. The transport of the emergency cooling water out of the upper plenum is effected simply by natural convection.

A comprehensive survey over the mass transport in the primary circuit and the resulting temperature difference between primary and secondary side of the steam generator, as measured in PKL, is given in figure 3 for the whole area of imaginable fluid dynamic conditions in the primary system.

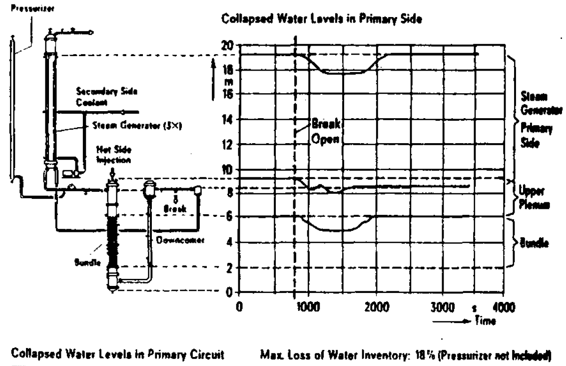

Fig.2: Water level in primary system according to PKL test, small leak (= 80 cm²) hot side injection

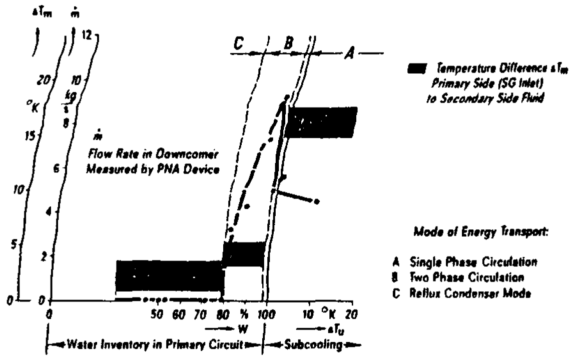

Fig.3: Mass flow and temperature difference in primary loop with small leak (PKL test)

Conditions can be single-phase free convection with subcooled water, the formation of vapour-liquid-mixtures in the region between core and steam generator up to the so-called reflux-boiler-condenser-mode in which vapour is rising out of the core, flows to the steam generator, condenses there, and the water comes back due to gravity on the same way. One recognizes that the evaporation augments the heat transport between core and steam generator, which can be seen in figure 3 from the reduction of the temperature difference between primary and secondary side of the steam generator. The driving force of the bubbles produced in the core at first improves the natural circulation via the steam generator, until finally with too high void a reflux of the condensed water in the rising part of the U-tubes of the steam generator occurs, without the necessity of a natural convection in the whole loop, as discussed before. The PKL-test-rig represents the same elevations as a 1300-MW pressurized water reactor, so that the natural convections in the reactor and in the experimental plant are identical. The cross section of all primary components is reduced by a factor of 134 with respect to the real plant. Scaling the experimental results to the reactor one finds that 100 kg/s vapour mass flow is enough to transfer the heat out of the core without fuel element damage. The temperature of the cladding is near to the saturation temperature of the water in the two-phase mixture.

One can now argue that with the emergency cooling water also nitrogen dissolved there can come into the primary loop which goes out of solution if the water is heated to saturation temperature. Also damaged fuel rods increase the content of non-condensible gases in the vapour and, finally, one has to take in account the hydrogen formed in a zirkon-water-reaction. As well known from chemical engineering non-condensible gases deteriorate the heat transfer by condensing. Tests with non-condensible gases were also made in the PKL-test-rig and the results are summarized in figure 4 /7/.

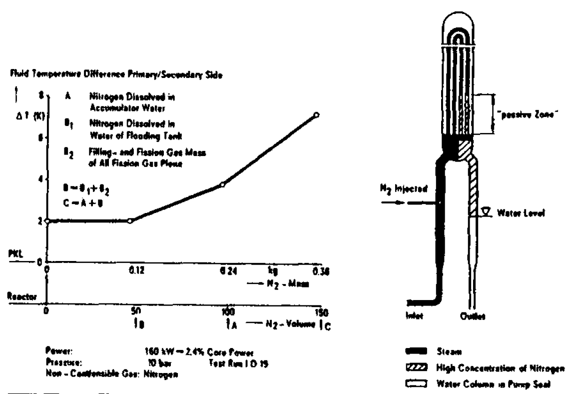

Fig.4: Influence of non-condensible gas on to the temperature difference in the steam generator for driving the heat transfer

The non-condensible gas is transported mainly to the descending branch of the U-tube-steam generator and the condensation occurs almost exclusively in the rising part. The heat transfer conditions certainly are worse compared to pure vapour, however, are still high enough to transport the heat produced in the core without fuel element damage. Whereas with pure vapour, the necessary driving temperature difference between primary and secondary side of the steam generators is in the order of 2 K, it has to become larger by a factor of 2 if all nitrogen contained in the emergency cooling water becomes free and is, finally, rising three times if the filling and fission gas from the fuel elements is added. However, even under very unfavorable conditions the temperature difference between primary and secondary side, which is necessary to transport the energy, would be only in the order of 6-10 K. This means, it is smaller than in normal operation.

However, even with very small leaks at which the incident sequence is much slower than with the 80 cm² leak discussed above, the question is rising, which consequences the partial or part-time failure of emergency core cooling systems may have. It has to be noted that with small leaks the safeguard system acts later and the cooling measures, therefore, also start later, because the thermodynamic and fluiddynamic situation is changing slower. Signals activating the emergency core cooling systems are

- lowering of the pressure in the primary system below 110 bar,
- falling of the water level in the pressurizer below 2,28 m or
- rising of the pressure in the containment above 1,03 bar.

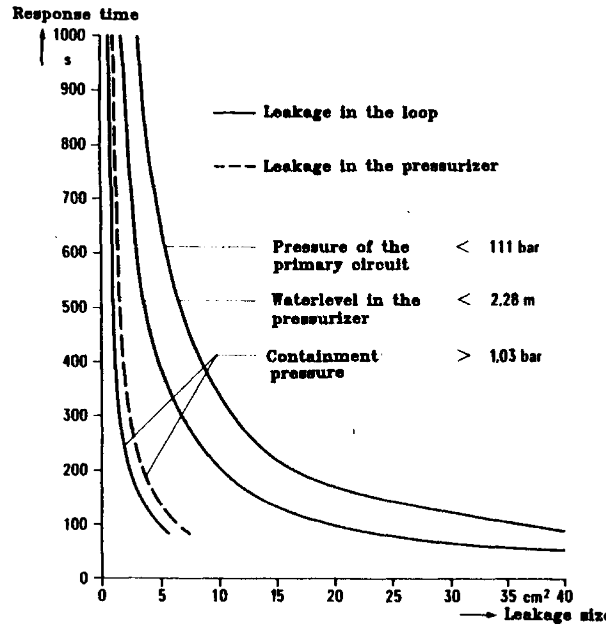

Fig.5: Response time of safety signals of a 1300 MW PWR

Fig.5 gives a rough information which time, depending on the size and the location of the leakage, is passing until each of these signals becomes active. Very small leakages - smaller than 5 cm² - do not cause an emergency cooling signal for very long time. For all that the cooling of the core is not endangered because the normal make-up system of the plant is good enough to compensate the water losses through the leakage, and the heat transport from the core to the steam generator occurs as in normal operation.

Best estimate ECC analysis for hypothetical fail combinations of ECC injection systems on prim.side (sec.side blow-down 100 K/h)

SMALL LEAK

leak size	system availability			max. cladding temperature
	high pressure inject. pump	accumulator	low pressure heat removal pump	
12 cm²	-	1	1	1200°C
25 cm²	-	3	1	1200°C
100 cm²	1	-	1	1200°C
12 cm²	1 1 h delayed	-	1	1200°C
25 cm²	1 1 h delayed	-	1	1200°C
50 cm²	1 20 min delayed	-	1	1200°C
100 cm²	1 10 min delayed	-	1	1200°C

Tab.3: Tolerable delay and partial failure of safety systems with small leaks

These small leakages would not need the injection of emergency core cooling water via the high pressure pumps at all in order to keep the cladding temperatures below 1200°C, as can be seen from the upper part of Tab.3. In the high pressure situation the heat transport via free convection to the steam generator is sufficient until finally the accumulators inject water into the core and for a long time the low pressure pumps take over the heat transport. With 100 cm² leakage area one high pressure safety injection pump is sufficient to avoid core damages. If one finally supposes that the accumulators - in spite of their passive operation - are not available, and three of the existing high pressure pumps fail also, even the last high pressure pump could start up to one hour later if the leakage is not larger than 25 cm². This tolerable time for the pump action delay is certainly decreasing with increasing leak area down to about 10 min, and with leakages larger than 100 cm² than again the low pressure emergency cooling pump takes over the cooling of the core. The situation is then similar to a large leak, which was discussed before.

With the incident sequences discussed up to now - i.e. with reduced availability of high pressure and low pressure emergency cooling pumps - it was assumed that the blow-down of the secondary side of the steam generator is started immediately after detecting the leak and occurs with 100 K/h. One can also insinuate a certain delay for activating the blow-down process, for example caused by a closing of the steam insulation valves of the secondary side, which separate the steam generator from the turbine. It then takes several minutes until the depressurization valves can work again. Also this situation would not be too serious, as figure 6 shows.

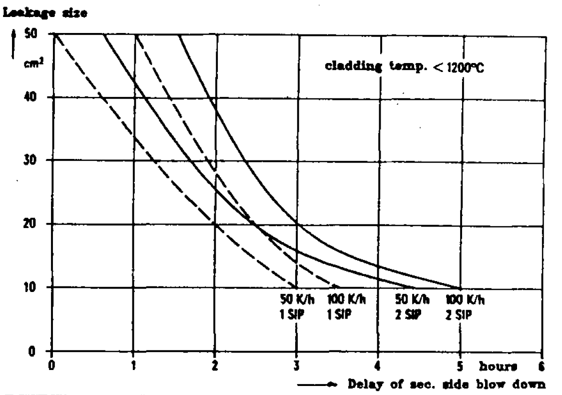

Fig.6: Emergency cooling analysis in case of reduced system availability and delayed sec. side blow-down (1300 MW PWR)

Even with only one high pressure emergency pump available this blow-down process could start up to 3,5 hours later, depending on the leak size, however, presumed that the cooling down velocity due to depressurization of the secondary side is not smaller than 100 K/h. But also with smaller coolin down velocities - for example 50 K/h - the is still enough time for activating the blow-down process. The availability of 2 high pressure emergency cooling injection pumps results in corresponding longer tole able times.

3.2 Measures to avoid core failure

As mentioned, risk studies /1,2/ showed t small leaks or also transients without a leak in the primary system - for example the case of loss of electrical power - giv the greatest contribution to the danger of a core failure and, with this, of a core melt down. Characteristic for all these in cidents is that the incident sequences pro ceed slowly and that by this time is avail able for safety directive measures activat by the operator. Prior condition for a saf action, however, is the reliable informati to the operator about the fluid- and therm dynamic situation in the primary loop and especially in the core. The only essential situation for the coolability of the fuel rods is the water level in the core. There fore, the knowledge of the water level in the core under incident condition has a ke position for initiating proper emergency- and protection-measures. The development o a reliable and precise water level indicat should, therefore, have much higher prior than intentions for future use of core me mitigation devices.

However, it seems to be also important to bring the risk analysis to the newest stat of knowledge about the thermohydraulic behaviour in the primary loops during an incident, because in the risk analysis avail able at present, core melt down is suppose already if the limits given in the licenci guidelines - i.e. 1200°C - are just excee By this the priority of incident signals and safeguard measures may be not correctl assessed in their relative order and small disturbances, which could be readjusted by the usual controlling devices, may unneces arily cause a scram or an emergency coolin signal which makes the conditions for the core more difficult and incorporates the danger of following failures.

3.3 Failure and desintegration of the cor

If one postulates a failure of emergency- and safeguard-systems beyond the extent di cussed up to now, the fuel rods will be da maged. At first a ballooning and bursting of the cladding will occur. Also under th situation the deterioration of the coolability was overestimated in the past. The conditions for bursting and ballooning of the cladding are well known today. Larger stretching before bursting only occurs if the temperature over the circumference of the cladding is almost completely uniform, and also under these very unfavorable and physically unrealistic conditions maximum stretching until bursting of 60% was observed in the Nuclear Research Center of Karlsruhe. In reality the temperature ove the circumference of the cladding is not uniform at all due to thermohydraulic reasons, which reduces the stretching /8/. In these tests it was found that even under unfavorable cooling conditions the ballooning of the cladding was not uniform versus the rod circumference, but one side orientated and it furthermore pointed out that the ballooned areas did not reach over a longer axial portion. The maximum reduction of the cooling channel by this ballooning was 52%.

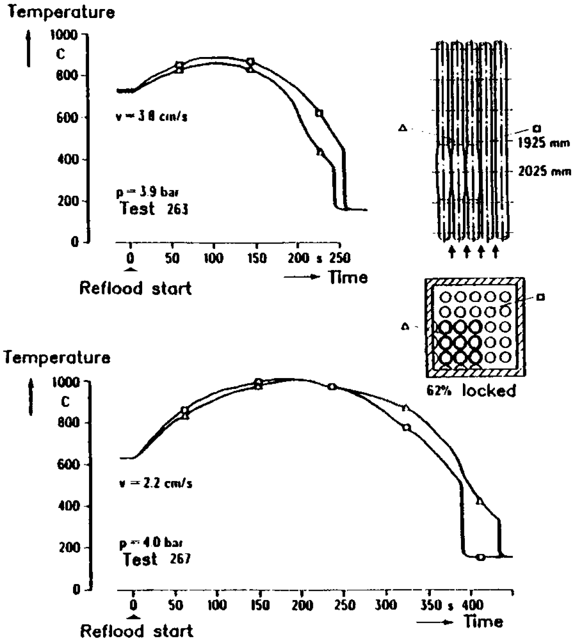

Fig.7: FEBA - Reflood test: Temperature of cladding in a partially locked bundle

These local reductions in the cooling cross section do not essentially impair the coolability of the rod, even if they would be uniform in azimutal direction, as demonstrated in figure 7 /8/.

4. Present and future research on core melt accidents

If one finally assumes that all emergency core cooling systems fail and that they can not be reactivated in time, a core melt would occur. Based on theoretical and experimental research, several steps and sequences are distinguished in the core melt down process. A synoptic analysis about the status of knowledge and the work in progress is performed in the German risk study, phase B, which is under way. Special emphasis is given there to
- the thermohydraulic of the primary system until dry-out of the core,
- accident sequences until the pressure vessel fails,
- sequences after the pressure vessel failure,
- temperature and pressure in the containment atmosphere,
- possibilities for violent chemical reactions between hydrogen and oxygen.

A failure of the containment due to overpressurization caused by steam production is to be expected not earlier than in the phase of the concrete-melt-interaction. This is valid for a core melt with small leaks (high pressures) as well as with large leaks (low pressures). The temporal pressure rise in the containment is strongly depending on the vapour production out of the concrete. The pressure at which the containment would fail is in the order of 8,5 bar.This pressure would be reached according to most recent calculations within a time of 3,5 to 4,5 days after the core melt process started. This result is independent from the causing event, i.e. large break, small leak or loss of electrical power. The temporal course of the pressure in the containment, as calculated on the basis of most recent research results, is shown in figure 8.

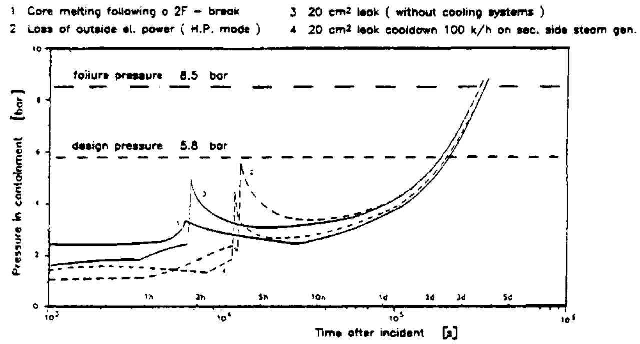

Fig.8: Temporal course of pressure in the containment

The penetration of the containment foundation due to indirection between the melt and the concrete before the containment would be over-pressurized, can be excluded with high certainty. The concrete penetration is a very slow process and it is not excluded that the erosion stops before the concrete would be penetrated. This is especially valid for PWR's with a basement layout against earthquakes and airplane crash down.

The failure of the pressure vessel under high pressure (small leak or loss of electrical power) can result in a steam spike in the containment, as shown in figure 8, in the period between 1,5 and 5 h. The reasons for that are, that the primary water, up to this moment still being in the primary system, is entering very rapidly the containment and that the water now released out of the accumulators comes in contact with the hot and partially desintegrated core, which gives an additional steam production. The largest pressure peak is to be expected for the case of loss of electrical power, however, even under pessimistic assumptions the containment pressure would not exceed the lay-out value. This sudden depressurization of the primary system can also result in a pressure peak in the shielding cavity, which may cause a local destruction of the biological shield.

Special attention has to be paid to the hydrogen behaviour. Assuming that there would be a homogeneous mixed atmosphere in the containment, no situation could occur which would enable a detonative H_2-combustion. For a deflagrative conbustion the mixture would become inflammable at the end of the core melt period earliest. A deflagration at this time could rise the pressure in the containment up to the lay-out pressure without probably causing a failure of the containment. After one day the production of additional inert gas - especially vapour - may be so high that a deflagration possibly could no longer occur. H_2-reactions are more dangerous with large leaks than with small ones. Reasons for this are that the H_2-production is smaller before the reactor pressure vessel fails and that the water evaporation in the sink starts earlier with small leaks. However, these results of risk studies and risk calculations are not yet secured. Especially a homogeneous mixture in the containment cannot be guaranteed.

From these deliberations one can set priorities for future research on core melt accidents and this was also done in the FRG. Emphasis of investigations in severe fuel damage research is given to
- hydrogen production (oxidation of ZRY and stainless steel and indirection between ZRY and UO_2),
- fuel rod behaviour,
- long term coolability of severely damaged core,
- fission gas release at high temperatures,
- H_2-behaviour and distribution in the containment and
- melt-concrete interaction.

Objects of these studies are to investigate the relevant physical and chemical phenomena, which enables us to develop better computer models and to quantify the safety margins existing in the safety systems of nuclear power plants. At present there are no intentions in the German Reactor Safety Commission to ask for new safety equipments especially designed against core melt down. The German RSK endeavoured to follow up a well balanced concept in their safety deliberations and to take care for a proportion in their safety related recommendations. This is, for example, documented in the RSK guidelines which are mainly on deterministic deliberations. If in the future probabilistic deliberations should become relevant in licencing discussions, one should - at least according to my opinion - remember in a higher degree as up to now the well-proved juridical principle of the "reasonability of the measures". Conclusions from risk analysis concerning safety measures against hypothetical accidents could else more impair the safety than to improve it.

Literature

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