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Cladding Deformation and Emergency Core Cooling of a Pressurized Water Reactor in a LOCA

Summary Description of the REBEKA Program

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Cladding Deformation and Emergency Core Cooling of a Pressurized Water Reactor in a LOCA - Summary Description of the REBEKA Program

ABSTRACT

The report summarizes significant results of the REBEKA program on the interaction between cladding deformation and emergency core cooling in a LOCA. The conclusion reached is that the major mechanisms have been sufficiently investigated in order to provide a reliable data base for safety assessments. All test data indicate that the coolability of deformed fuel elements of a PWR can be maintained in a LOCA.

Hüllrohrverformung und Kernnotkühlung eines Druckwasserreaktors beim Kühlmittelverluststörfall - Zusammenfassung des REBEKA Programms

ZUSAMMENFASSUNG

Der Bericht faßt die wichtigsten Ergebnisse des REBEKA-Programms über die Wechselwirkung zwischen Hüllrohrverformung und Kernnotkühlung beim Kühlmittelverluststörfall zusammen. Die durchgeführten Untersuchungen haben zu einem ausreichenden Verständnis und zur Quantifizierung der entscheidenden Mechanismen geführt und stellen eine zuverlässige Basis für Sicherheitsbewertungen dar. Alle Ergebnisse zeigen, daß keine Beeinträchtigung der Kühlbarkeit deformierter Brennelemente beim Kühlmittelverluststörfall zu erwarten ist.

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1. INTRODUCTION

Within the framework of the licensing procedure under the Atomic Energy Act evidence must be produced that the impacts of all conceivable pipe ruptures in the primary loop of a pressurized water reactor (PWR) can be controlled. The double-ended break of the main coolant line between the main coolant pump and the reactor pressure vessel is presently considered to constitute the design basis for the emergency core cooling system (ECCS). Upon rupture of the reactor coolant line the reactor is shut down. However, as the production of decay heat continues, reliable long-term cooling of the reactor core is required.

After depressurization and evacuation of the reactor pressure vessel emergency cooling systems supply the reactor core with the emergency cooling water kept in the accumulators and the flooding tanks. Cooling of the fuel elements is temporarily deteriorated until emergency core cooling becomes fully effective. In that time interval, some fuel rod claddings attain temperatures at which they balloon or burst under the impact of the internal overpressure. Figure 1 shows as an example the load on a fuel rod cladding in a loss-of-coolant accident (LOCA) as predicted by conservative evaluation models for a German PWR.

In the LOCA analysis, the number of ruptured fuel rods, the extent and distribution of cladding deformations and their influence on coolability must be predict-

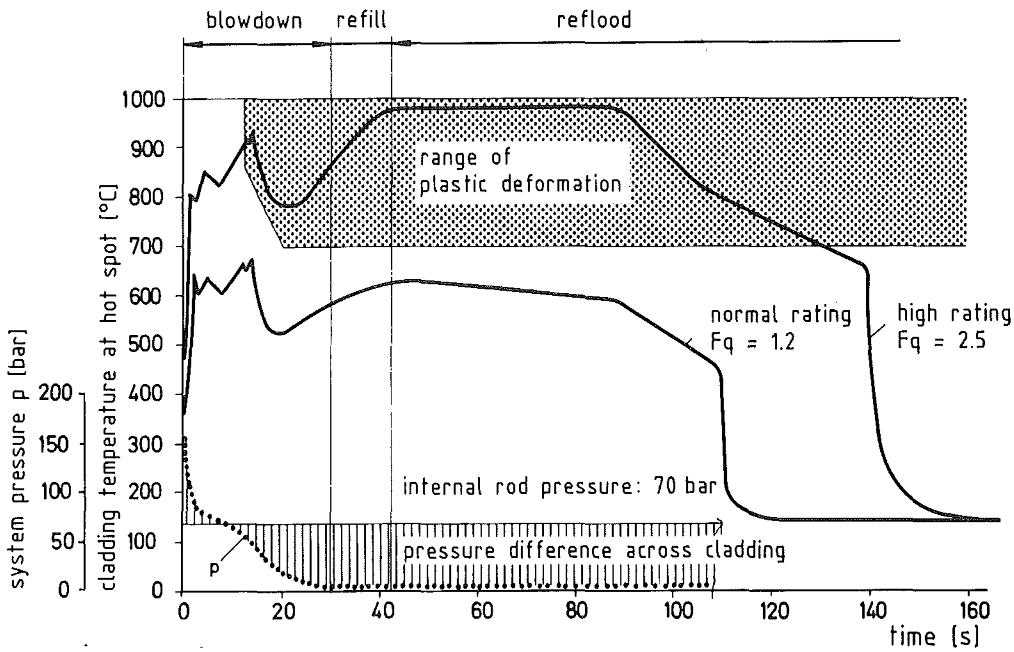


Fig. 1: **Fuel rod cladding load in a double-ended cold leg break LOCA**

ed. The REBEKA-program (REactor typical Bundle Experiment KArlsruhe) of KfK furnished a broad data base allowing to judge cladding deformation and emergency core cooling. The primary purpose of the program was to investigate the interaction between thermal-hydraulics and cladding tube deformation.

2. EXPERIMENTAL

The test program was performed in several consecutive test series from single-rod tests conducted in steam atmosphere to bundle tests comprising 7 x 7 rods under flooding conditions. The range of test parameters was chosen to cover the emergency core cooling conditions of a German PWR.

2.1 Fuel Rod Simulator

An essential requirement for the experiments was the development of an electrically heated fuel rod simulator which models the nuclear fuel rod, in particular its thermal and deformation behavior during reflooding. Figure 2 shows the schematic design and the axial power distribution of the REBEKA fuel rod simulator. Its excellent thermal simulation compared to a nuclear fuel rod has been proved in extensive experimental work [1].

The Zircaloy-4 cladding tubes having an outer diameter of 10.75 mm and an inner diameter of 9.3 mm were cold worked and stress relieved.

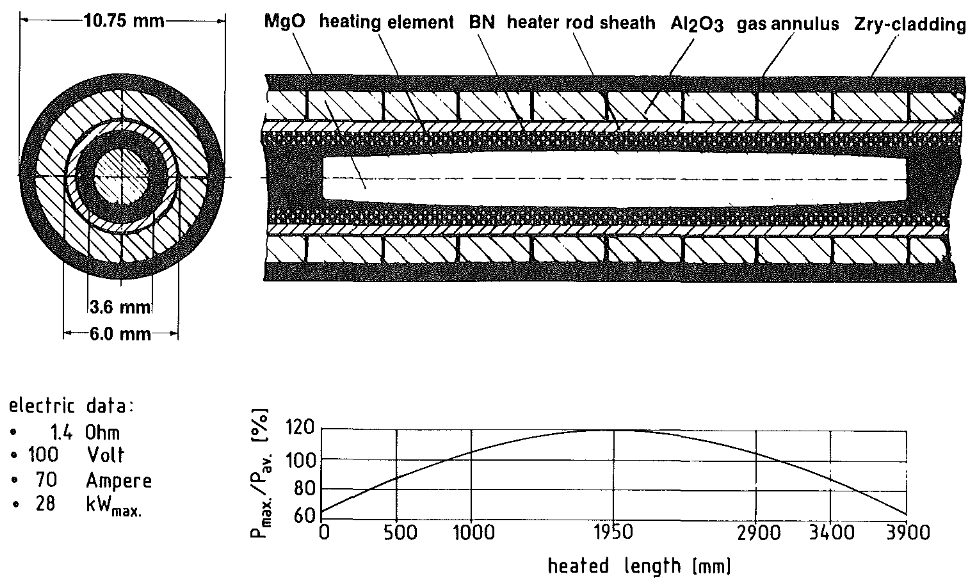


Fig. 2: REBEKA fuel rod simulator

2.2 Single-rod Test Rig

Figure 3 shows schematically the test rig used for the single-rod deformation and burst tests. Fuel rod simulators of 325 mm heated length with constant axial power are used. For internal pressurization of the cladding tubes helium is used. A heatable shroud tube acting simultaneously as a steam guiding pipe can be used to minimize the temperature differences along the circumference of the cladding tubes or to cause temperature differences, if the shroud is more or less unheated in order to simulate external heat transfer. To record time dependent ballooning of the cladding, an X-ray system with film recording is used. The evaluation of the individual frames of the film provides the information on the time dependent development of strain which is important to elaborate a computer model for calculating the cladding deformation.

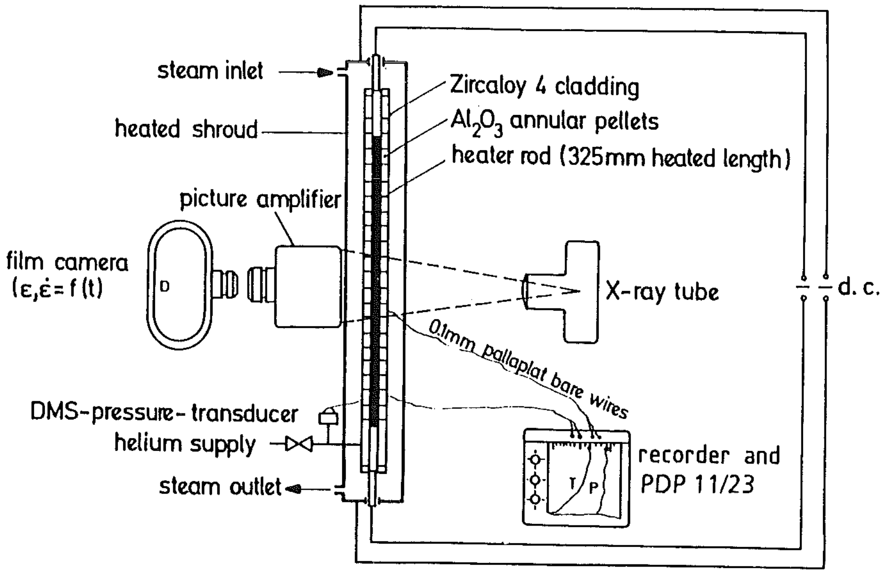


Fig. 3: **Single-rod test rig**

The cladding temperatures are measured by 0.1 mm pallaplat bare-wire thermocouples spot welded on the outer surface of the cladding.

The test parameters covered a broad range: 10 to 140 bar for the internal rod pressure and 1 to 30 K/s for the heating rate. The test atmosphere was almost stagnant steam at atmospheric pressure and 200°C.

2.3 Bundle Test Loop

Figure 4 illustrates schematically the REBEKA test loop used for the reflooding bundle tests. Figure 5 is a photo of the 49 rod bundle. Bundle configurations in a 5×5 and 7×7 square array with a pitch to diameter ratio of 1.33 and original grid spacers were used. The fuel rod simulators had a heated length of 3900 mm and an axial power profile with a peaking factor of 1.19

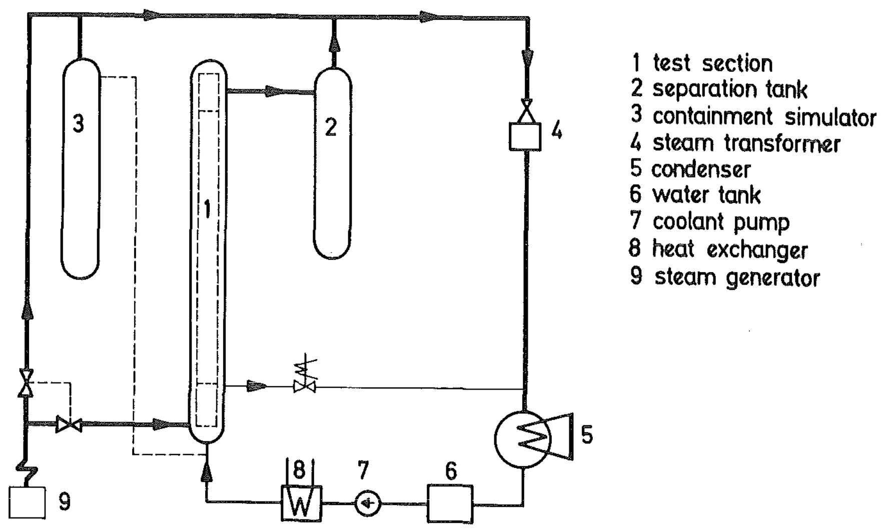


Fig. 4: Bundle test loop

An essential feature of the loop is that by providing representative emergency cooling conditions during reflooding and by use of electrically heated fuel rod simulators of high simulation quality, the temperature and pressure transients of the cladding tubes are established automatically in a representative way without controlling the power supply.

Inconel-sheathed 0.5 mm outside diameter NiCr/Ni thermocouples with platinum sleeves at the tip, spot welded onto the outer surface of the Zircaloy claddings are used to measure the temperature history of the Zircaloy cladding. In addition to the outer cladding thermocouples, all internal heater rods of 6.0 mm outer diameter are equipped with sheathed 0.36 mm outer diameter NiCr/Ni thermocouples embedded in grooves.

The 49-rod bundle is equipped with 49 pressure transducers to record the internal rod pressures and with about 150 thermocouples to record the heater rod and cladding temperatures.

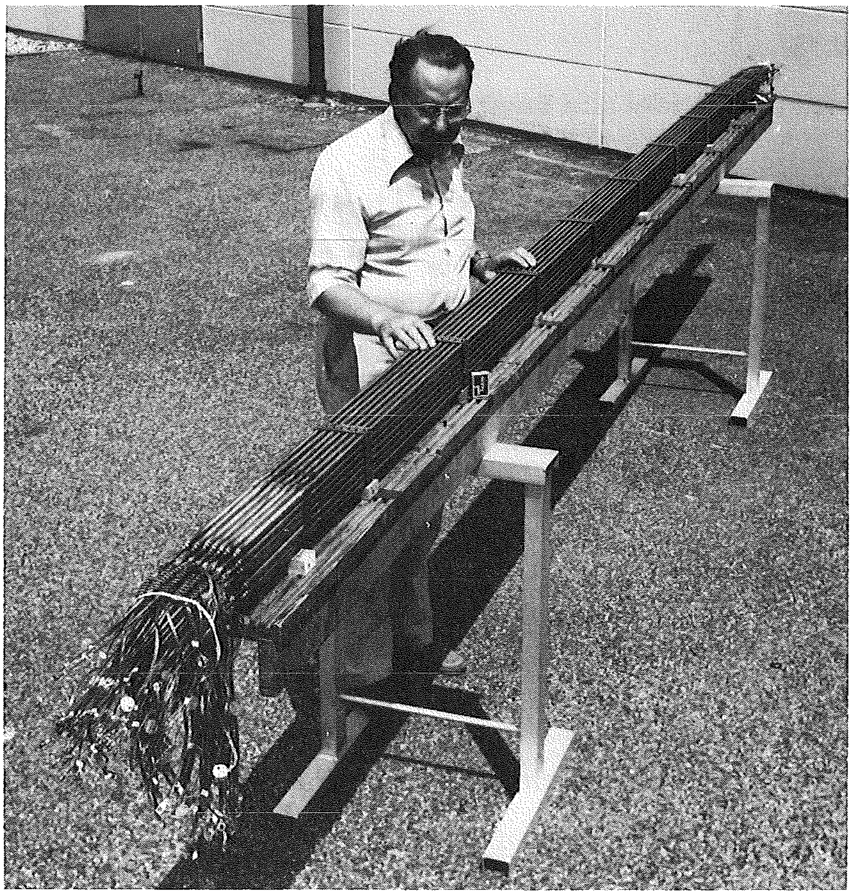


Fig. 5: 49-rod test bundle

All measurements are recorded during the tests by a fast computer-controlled data acquisition system using a PDP 11/23 computer. This system scans each channel every 0.1 s at a frequency of 10 kHz. - The data acquisition system has been used for both the single-rod and the bundle tests.

In the bundle reflooding and deformation tests, a heat transfer coefficient of 30 W / (m^2K) during the refilling phase was simulated by a steam flow of about 2 m/s. Reflooding was initiated from the bottom by forced feed at a cold flooding rate of 3 cm/s, a feed water temperature of 130 °C, and a system pressure of 4 bar. The fuel rod simulators were heated by a maximum decay heat power of 20 W/cm which is typical of an average rated fuel rod in the core.

The internal rod pressure was simulated by pressurized helium and adjusted to 70 bar at the beginning of the heat-up phase in a closed rod volume. This internal pressure is representative of the conditions prevailing in German PWRs and results in maximum cladding tube deformations in a LOCA.

3. RESULTS

3.1 Deformation Model and Burst Criterion

Based on the test results of the single-rod burst tests, a computer model was developed to calculate the deformation process and the burst data of Zircaloy cladding tubes for known temperature and pressure histories [2]. It is assumed in the model that the process of deformation up to burst of thin-walled, internally pressurized Zircaloy cladding tubes can be calculated from the steady-state (secondary) creep equation (NORTON equation) including the effects of oxidation. According to the burst criterion the time of burst is reached when the actual local stress equals the limiting burst stress. Based on the single-rod burst tests, it is assumed that the burst stress depends on the temperature and oxygen concentration of Zircaloy. With the histories of temperature and pressure known, the actual data of strain, stress, and oxygen content can be calculated by integration of the creep equation and correlation of the oxidation kinetics. Once the time of burst has been calculated, all burst data, i.e. strain, stress, temperature, pressure, oxygen content are determined.

Figure 6 illustrates and summarizes schematically the approach adopted to develop the deformation model and the burst criterion. The results of the model are plotted in the following diagrams as curves and compared with the experimental data.

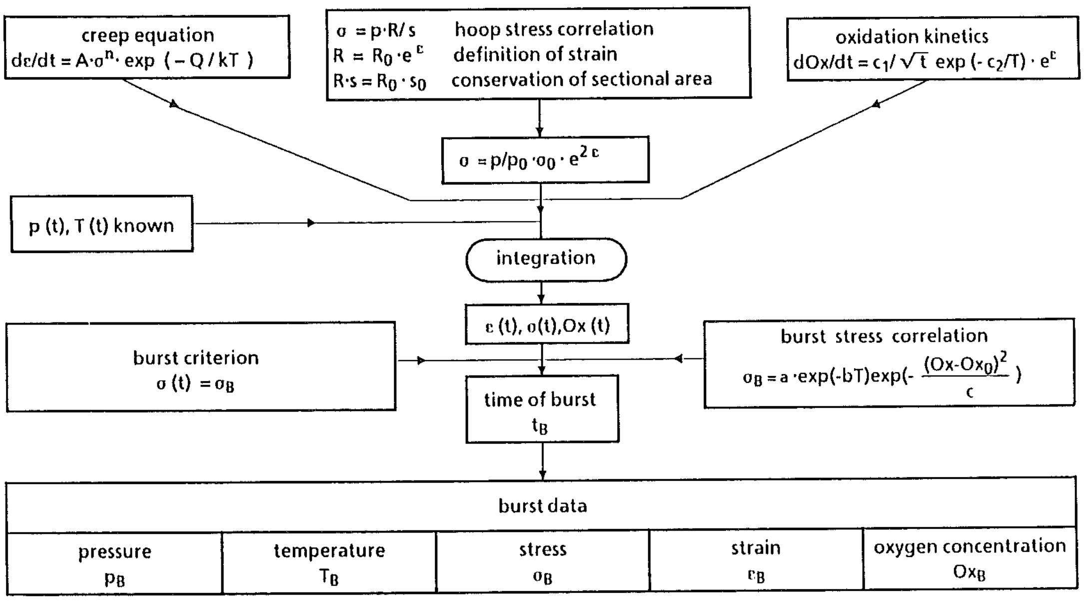


Fig. 6: REBEKA deformation model and burst criterion

3.2 Burst Data

In order to record the consequences both of a large break and of small breaks (small differential pressure) and to take account of developments resulting in an increase in target burnup (high differential pressure), the deformation and burst behavior was investigated for a wide pressure range in single rod experiments. However, the investigations concentrated on the range of burst pressures of about 50 to 70 bar which must be supposed to act in a large break design basis accident of a German PWR.

3.2.1 Burst temperature

Experimental data of the burst temperature among other factors provide an important basis for the prediction of the number of burst cladding tubes and for the fission product release resulting from them.

Figure 7 shows the burst temperature versus the burst pressure with the heating rate as the parameter. The calculated curves describe the results of the single-rod

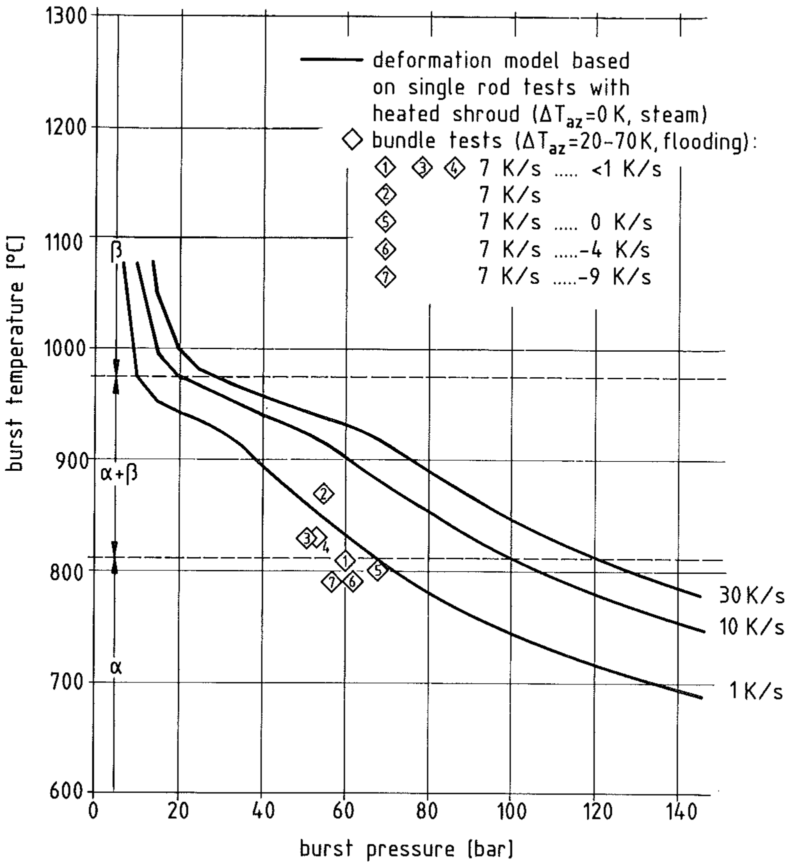


Fig. 7: Burst temperature vs. burst pressure

tests. At the same heating rate a higher rod internal pressure causes the burst temperature to become lower. The results show a significant influence of the heating rate on the burst temperature, i.e., high heating rates lead to higher burst temperatures than low heating rates. The results of the bundle tests are in good agreement with the single-rod test results and show also the burst temperature depending on the heating rate.

With this information the number of burst fuel rods in a LOCA can be determined with adequate accuracy if the temperature and pressure development of the fuel rods is known. Accordingly, with the inner pressures and burnups typical of a German PWR, the cladding tubes will fail through burst when temperatures of about 800°C are attained.

3.2.2 Circumferential burst strain

The size of circumferential burst strain of the Zircaloy cladding tubes determines among others decisively the coolant channel blockage and the coolability in the fuel element.

Figure 8 shows the circumferential burst strain versus the burst temperature with the heating rate as the parameter. The calculated values represented by curves describe the circumferential burst strains measured in the single-rod tests on the cladding tube circumference at uniform temperature. These idealized conditions were specifically provided by a heated shroud surrounding the fuel rod simulator.

The averaged values from the bundle tests entered in the diagram indicate a significant reduction in the circumferential burst strains to values around 50%. This limitation is due to temperature differences on the cladding tube circumference.

Under representative thermal-hydraulic boundary conditions of a LOCA as simulated in the bundle tests, heat flows from the pellet across the gas filled gap to the cladding tube and the coolant. Tolerances in the dimensions of the pellets and cladding tubes as well as eccentricities of the pellets in the cladding tube lead to differences in gap widths along the cladding tube circumference and, consequently, to differences in heat transfer across the gap between the pellets and the cladding tube. In case of external cooling, this results in temperature differences at the cladding tube circumference (azimuthal temperature differences).

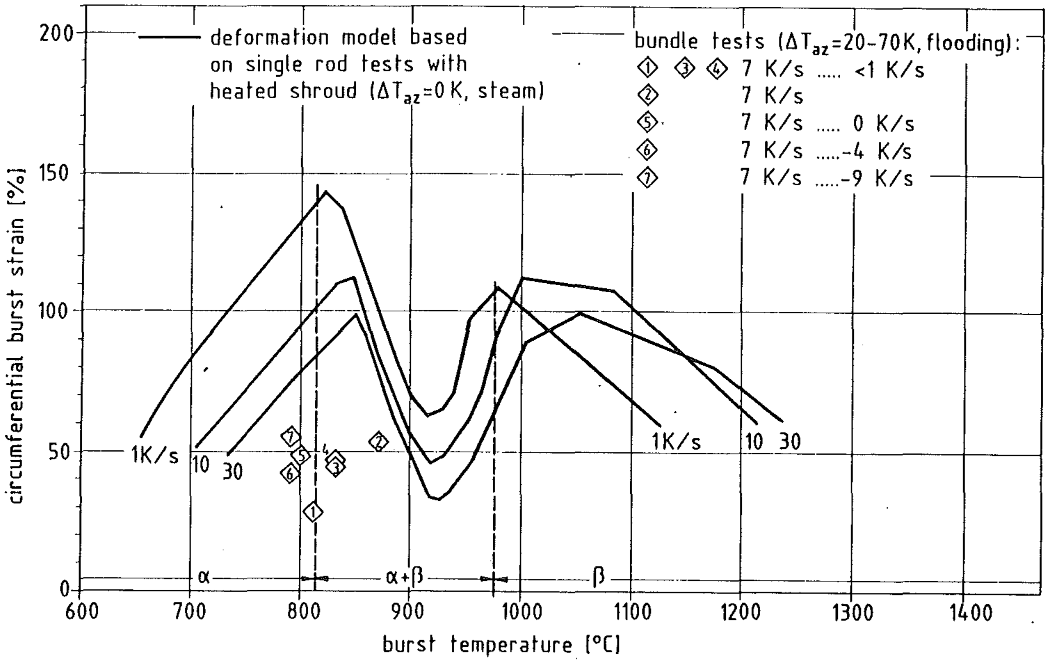


Fig. 8: **Burst strain vs. burst temperature**

In the single-rod tests in which temperature differences developed on the cladding tube circumference due to the unheated shroud, it has been proved that in case of deformation of Zircaloy claddings in the α- and early (α + β)-phases of Zircaloy a systematic relationship exists between the circumferential burst strain and the azimuthal temperature difference: Small azimuthal temperature differences cause a relatively uniform reduction in cladding tube wall thickness on the circumference and give rise to relatively high circumferential burst strains; high azimuthal temperature differences result in a preferred reduction of wall thickness on the hot part of the cladding tube circumference and lead to relatively low circumferential burst strains.

Figure 9 shows quantitatively the dominant influence of the azimuthal temperature difference on the circumferential burst strain. These relationships can be explained by the anisotropic strain behavior of Zircaloy and the resulting bowing of the Zircaloy cladding tubes observed during deformation accompanied by azimuthal temperature differences [3]. Tube bowing represented in the figure produces the effect that the gap between the pellets and the cladding tube closes on the hot side and opens on the opposite cold side. This leads to increasing temperature differences during cladding tube deformation.

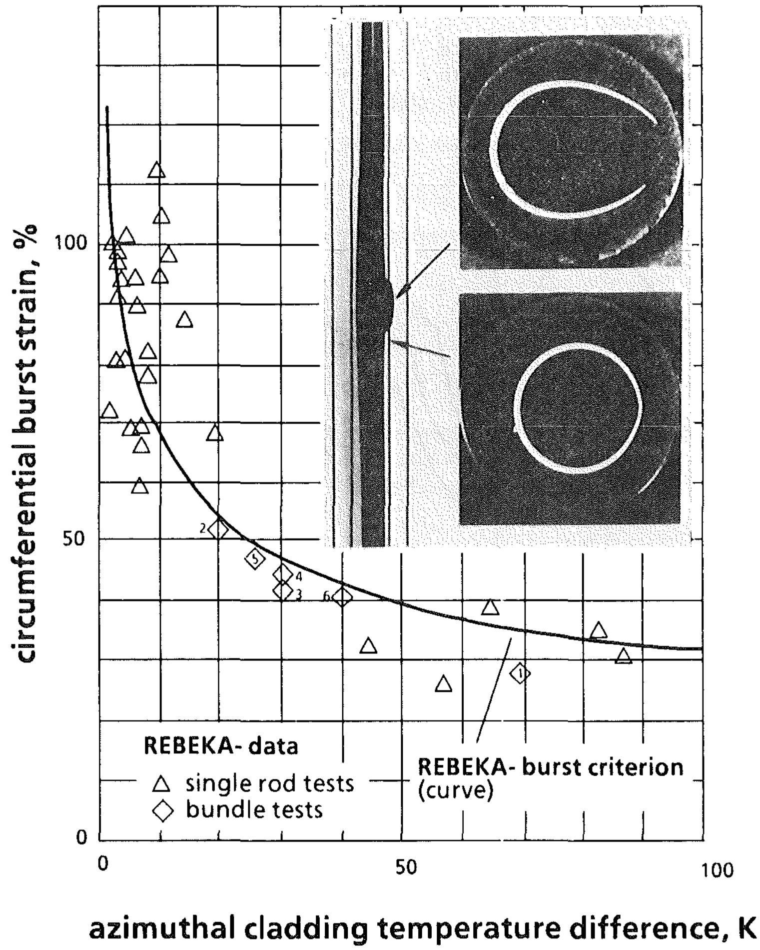


Fig. 9: Burst strain vs. azimuthal temperature difference

3.3 Influence of Heat Transfer on Cladding Deformation

It has been found that the circumferential burst strain of the Zircaloy cladding tubes becomes the smaller the higher the heat transfer from the cladding tube to the coolant is. This is attributable to tube bowing described above (see Fig. 9). As the hot cladding tube side contacts the heat source and the opposite cold side bends continuously away from the inner heat source, intensified external cooling gives rise to an enhancement of the differences of the azimuthal cladding tube temperatures and, as a result, a reduction in circumferential burst strain.

Figure 10 makes evident that bundle tests which are performed with very low heat transfer, for instance low steam cooling, will necessarily lead to relatively high circumferential burst strains. Relatively low circumferential burst strains [4], however, have been observed in bundle tests which are characterized by heat transfer coefficients higher than 50 W/(m^{2}K) during deformation. Such results are typical of the reflooding phase of a LOCA.

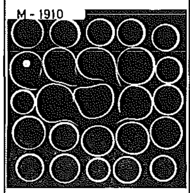
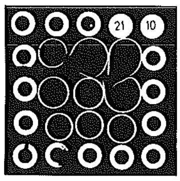
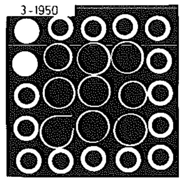
	REBEKA-M	REBEKA-2	REBEKA-3
cross-section at max. flow blockage			
fluid flow	stagnant steam	steam flow	two-phase flow
heat transfer coefficient [W/m²·K]	<10	~30	~30 ÷ 100
mean burst strain of inner 3x3 rods [%]	63	54	44

Fig. 10: **Influence of heat transfer on cladding deformation**

In all experiments performed under typical thermal-hydraulic conditions, mean azimuthal cladding tube temperature differences of about 30 K developed at the time of burst which limit the mean circumferential burst strain to values of about 50%.

3.4 Influence of Flow Direction on Coolant Channel Blockage

The coolant channel blockage caused by ballooned and burst cladding tubes in the fuel element depends on the extent of the cladding strain and its axial position. If the burst points are displaced over a rather large axial region, the coolant channel blockage is relatively low. However, in case that the burst points occur rather closely to each other, the resulting coolant channel blockage is greater for the same mean burst circumferential strain. As plastic deformation of Zircaloy cladding tubes responds very sensitively to the cladding tube temperature, the axial displacement of the burst points is determined crucially by the axial position of the temperature maximum of the individual cladding tubes.

The heat transfer between the rods and the mixture of steam and water droplets is effected almost exclusively by convection. As the heat transfer from the cladding tube wall to the steam is much more pronounced than the heat transfer from the steam to the water droplets, a thermodynamic non-equilibrium develops during the reflooding phase in two-phase flow which means that the steam is superheated along the coolant channel. In the bundle tests, steam temperature of up to about 600°C were measured which corresponds to a superheat of about 450 K. At the grid spacers, the incident water droplets are split into a population of smaller droplets so that on account of the larger droplet surface a more effective heat sink is produced for the highly superheated steam. Together with the enhanced turbulence downstream of each spacer, this leads to a reduction in steam and cladding tube temperatures. However, in the direction of flow up to the next spacer, the degree of superheat increases again which leads to the development of an axial temperature profile [5].

During a LOCA the direction of flow in the reactor core depends on the design and availability of the ECCS's and on their interaction with the primary loops. Besides local differences in flow and steam/water countercurrent flows, two characteristic and limiting flow directions exist in the reactor core in a combined hot/cold leg injection mode which is typical of a German PWR as regards cladding tube deformation and coolant channel blockage, i.e. flow reversal from the refill to the reflooding phases and unidirectional flow during the refill and reflooding phases.

Figure 11 illustrates the impacts of a flow reversal on the circumferential strain of the Zircaloy cladding tubes and the resulting coolant channel blockage. In the experiment (REBEKA 5), the rod bundle was passed by steam flow from top to bottom during the refill phase and from bottom to top during subsequent reflooding with water in order to simulate flow reversal. Therefore, during the refill phase, the maximum of cladding tube temperature initially moves downward towards the spacer below the mid-plane as a result of the downward directed steam flow. In the subsequent flooding phase, the temperature maximum is displaced in the direction of flow with the flooding time getting longer towards the spacer above mid-plane, i.e., the temperature maximum between two spacer positions develops at different axial positions with time. But due to inhomogeneities in the rod bundle resulting from locally differing rod powers and cooling, not all the rods are heated up uniformly which results in different burst times. In REBEKA 5, the burst time interval of the individual Zircaloy claddings was about 24 seconds. During that time interval, there was a shift in the temperature maximum which automatically led to an axial displacement of the burst points over a rather large range. It is evident from the figure that the burst points are spread

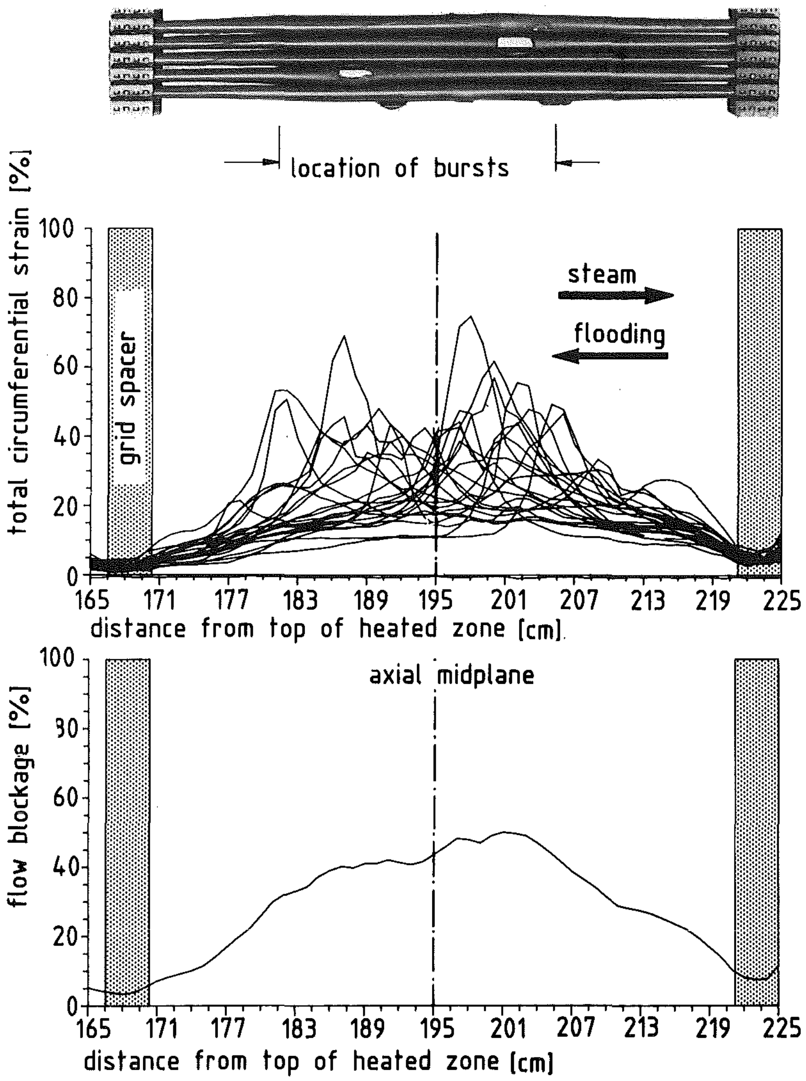


Fig. 11: **Cladding deformation and flow blockage under reversed flow (REBEKA 5)**

over some 24 cm of axial length around the mid-plane which gives rise to a relatively low maximum coolant channel blockage of 52%.

Figure 12 shows the deformation pattern for undirected flow in the rod bundle. In that experiment (REBEKA 6) the flow direction of the coolant from bottom to top was maintained during the refill and reflooding phases. Unlike in REBEKA 5, the temperature maximum was moved from the very beginning of the experiment towards the upper of the two medium spacers. After that temperature profile had developed during the refill phase, the temperature maximum continued to occur at approximately the same axial position. This led automatically to a local concentration of the burst points and, consequently, to a higher coolant channel blockage. The figure exhibits a pronounced displacement of the burst points in the direction of flow towards the upper spacer. The burst points are displaced

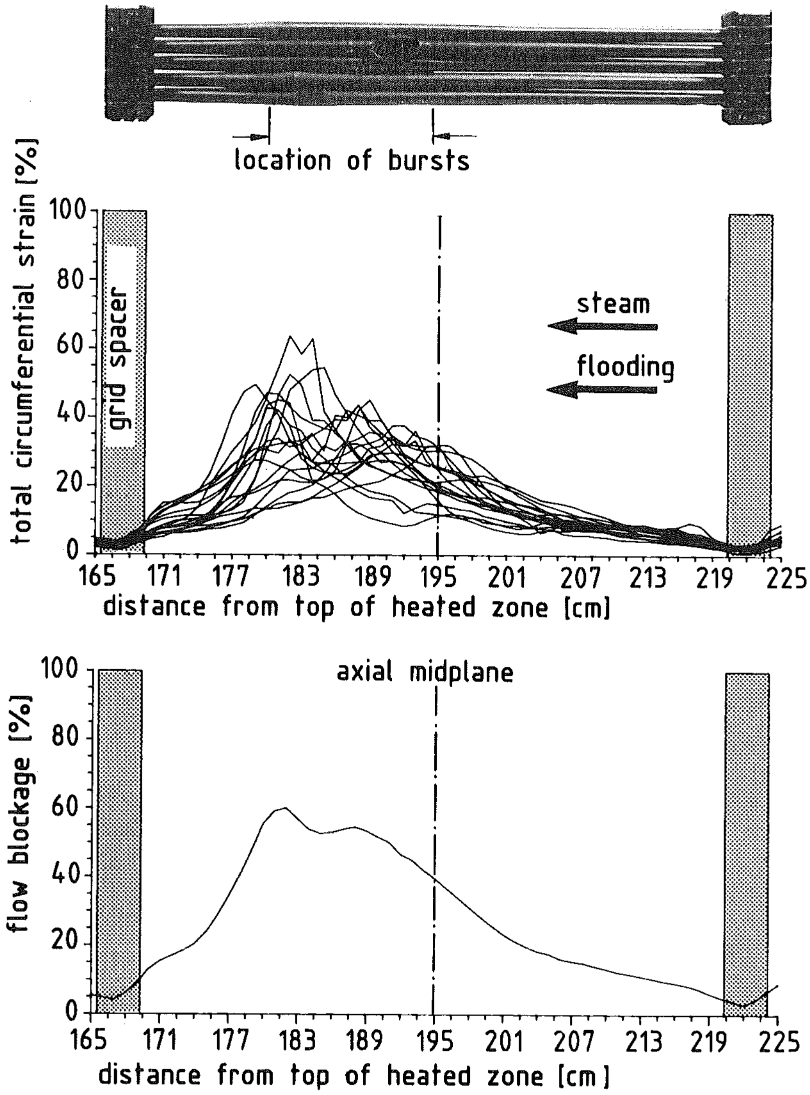


Fig. 12: Cladding deformation and flow blockage under unidirectional flow (REBEKA 6)

only over an axial zone of about 14 cm because the flow direction has been maintained which results in a greater coolant channel blockage of 60%.

In the REBEKA 7 experiment the flow direction was likewise maintained, but the cooling conditions during reflooding were set to produce maximum interaction between the deformation and the coolant and maximum coolant channel blockage. In that test the greatest coolant channel blockage to be expected under representative flooding conditions was about 70%.

3.5 Coolability of Deformed Rod Bundles

Flow blockages produced by ballooned fuel rod claddings change the cooling mechanism and cause two counteracting effects on the local reflood heat transfer:

- A flow bypass effect, which reduces the coolant mass flow in the blocked region and, consequently, decreases the local heat transfer.

- A flow blockage effect, which leads to water droplet dispersion and evaporation and turbulence enhancement, thus increasing local heat transfer.

It has been found in all REBEKA bundle tests with forced feed flooding that the effect of water droplet dispersion, which improves the heat transfer, overcompensates the degrading effect of mass flow reduction. Moreover, premature quenching and improved coolability due to burst cladding tubes has been demonstrated. Therefore, in all tests an almost instantaneous stabilization and reduction of the cladding temperature was evident right after the start of flooding and beginning of clad ballooning.

Figure 13 shows the cladding temperatures and internal rod pressures of the inner 25 rod of the test REBEKA 7 which exhibited the maximum conceivable coolant channel blockage of approx. 70%. The temperature plots are a proof of the excellent coolability of the deformed rod bundle. It is evident that shortly after the initiation of flooding a further increase in the cladding temperatures is stopped and the cooling effect of the two-phase mixture is very effective even in the upper elevation of 2100 mm represented in the diagram. The lower part of Fig. 14 shows the internal rod pressure of the fuel rod simulators. An instantaneous pressure drop to the system pressure of 4 bar indicates the time of burst of the individual cladding tubes. The diagram illustrates that soon after burst of the claddings quenching occurs. The maximum cladding temperatures reached in the test were only slightly higher than 800 °C, i.e. substantially below the upper limit of 1200°C set by the emergency core cooling criteria.

The REBEKA tests have shown that burst Zircaloy cladding tubes improve the coolability compared with non-deformed cladding tubes. In case of burst cladding tubes burst lips penetrate into the coolant channels and the helium simulating the fission gas escapes from the rod through the burst opening. Steam with its

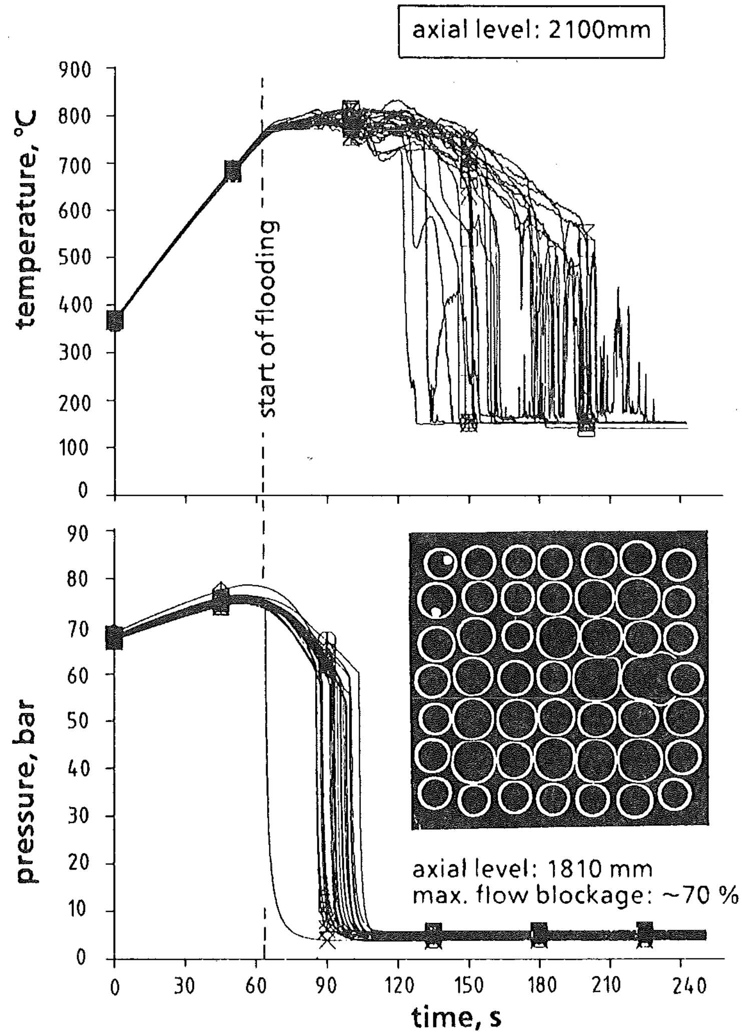


Fig. 13: Temperature and pressure transients of a 70 % blocked rod bundle (REBEKA 7)

much poorer heat conductivity enters the rod through the burst and decouples the cladding from its heat source. The two-phase coolant flow is now capable of cooling down the cladding very fast to the quench temperature at the point of burst. Two new secondary quench fronts propagate from the burst. One of them moves upwards, the other downwards opposed to the direction of flow. The quench front moving upwards reaches the upper rod end earlier than the regular front in a non-deformed fuel element.

For REBEKA 6 with a maximum coolant channel blockage of 60%, Fig. 14 shows temperature transients for Zircaloy cladding tubes measured at different axial positions between the two central grid spacers. It can be recognized that the quench front propagates faster in the direction of flow than in the opposite direction. The regular quench front of a non-deformed Zircaloy cladding attains the

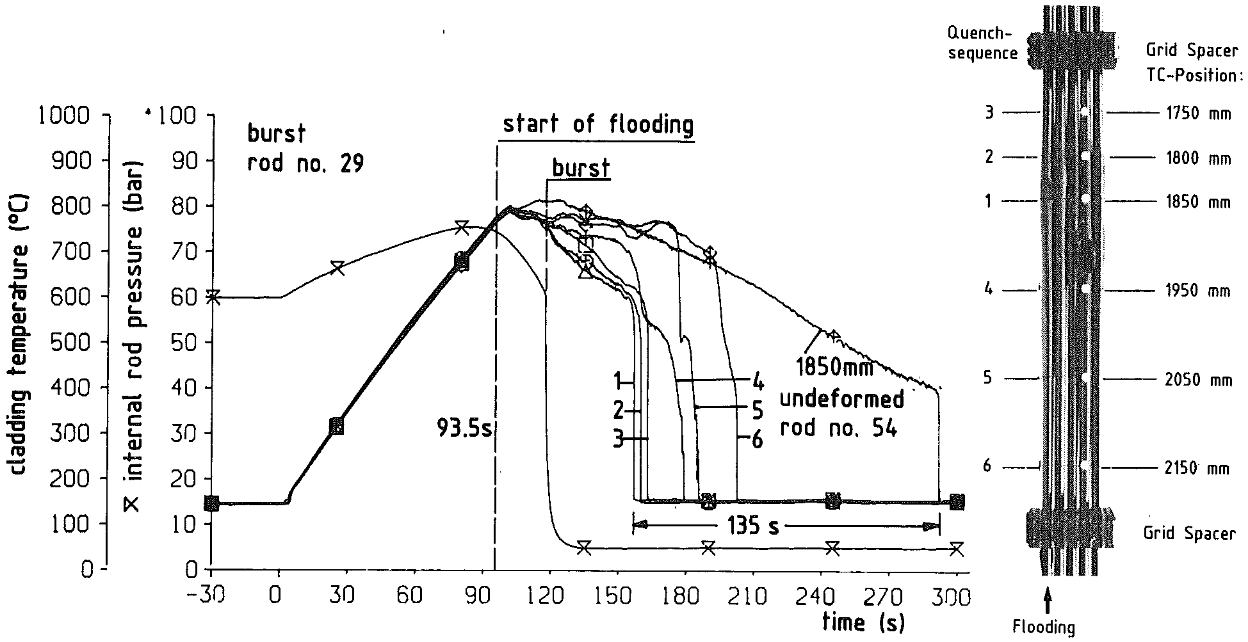


Fig. 14: Temperature transients and quenching behavior of a burst rod (REBEKA 6)

axial position of 1850 mm only 135 s later than the secondary quench front of the burst rod.

These results prove that the hot zones in a fuel element accommodating burst claddings have a very favorable cooling behavior.

4. SUMMARY AND CONCLUSION

The most important results of the REBEKA-program can be summarized as follows:

- The number of burst cladding tubes and their circumferential burst strain can be determined with sufficient accuracy if the temperature and pressure developments of the cladding are known.
- The circumferential burst strains of Zircaloy cladding tubes are kept relatively small due to temperature differences on the cladding circumference and the anisotropic strain behavior of Zircaloy.
- The cooling effect of the two-phase flow increases the temperature differences on the cladding tube circumference and limits in this way the mean circumferential burst strains to values of about 50%.

- A unidirectional flow through the rod bundle during the refill and reflooding phases causes cooling channel blockage of about 70% at the maximum.
- Burst cladding tubes generate secondary quench fronts which propagate relatively fast and lead to effective cooling.

All effects described underline that in a LOCA of a German PWR no impairment of the coolability must be expected to occur and that the safety margin is greater than predicted by most of the computer codes.

REFERENCES

[1] Erbacher, F.J., Ihle, P., Rust, K., Wiehr, K. "Temperature and Quenching Behavior of Undeformed, Ballooned and Burst Fuel Rods in a LOCA," Fifth International Meeting on Thermal Nuclear Reactor Safety, September 9-13, 1984, Karlsruhe, FRG; KfK-3880/1B /Dezember 1984), S. 516-24.

[2] Erbacher, F.J., Neitzel, H.J., Rosinger, H., Schmidt, H., Wiehr, K. "Burst Criterion of Zircaloy Fuel Claddings in a LOCA," ASTM Fifth International Conference on Zirconium in the Nuclear Industry, August 4-7, 1980, Boston, Mass., USA, ASTM 1982, S. 271-83 (ASTM Special Technical Publication 754).

[3] Erbacher, F.J. "Experimentelle Untersuchungen zur Hüllrohrdeformation und Kernnotkühlung eines Druckwasserreaktors beim Kühlmittelverluststörfall," doctoral thesis, Ruhr-Universität Bochum, März 1990.

[4] Wiehr, K. "REBEKA-Bündelversuche, Untersuchungen zur Wechselwirkung zwischen aufblähenden Zircaloyhüllen und einsetzender Kernnotkühlung, Abschlußbericht," KfK 4407, April 1988, Kernforschungszentrum Karlsruhe GmbH, Karlsruhe, FRG.

[5] Rust, K., Ihle, P., "Reflood Heat Transfer Tests for PWR Safety Evaluation," Proceedings of Thermophysics-90, September 25-28, 1990, Obninsk, USSR.