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Fuel Behavior under Loss-of-Coolant-Accident Conditions

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Abstract

The paper is a comprehensive summary of the main results of the KfK/PNS investigations on LWR fuel behavior under LOCA conditions.

These investigations were started in 1973 and will be finished in 1983.

It is shown that the dominant phenomena, such as the deformation and failure of the cladding, the high temperature steam oxidation, the interaction of the cladding with fuel and fission products, and the influence of thermohydraulics on the cladding deformation are well understood today. All results confirm that under LOCA conditions the coolability of the core is not questioned and the fission product release is well below license limits.

Kurzfassung

Brennstabverhalten bei Kühlmittelverluststörfällen

Der Vortrag gibt einen kurzgefaßten Überblick über die wesentlichen Ergebnisse der KfK/PNS-Untersuchungen zum LWR-Brennstabverhalten unter LOCA-Bedingungen.

Diese Untersuchungen wurden im Jahre 1973 begonnen und werden 1983 beendet.

Es wird gezeigt, daß die dominierenden Phänomene, wie z.B. die Deformation und das Versagen der Hüllrohre, die Hochtemperaturdampfoxidation, die Wechselwirkung der Zircaloy-Hülle mit Brennstoff und Spaltprodukten und der Einfluß der Thermohydraulik auf die Hüllrohrdeformation heute gut verstanden werden. Alle Ergebnisse belegen, daß unter LOCA-Bedingungen die Kühlbarkeit des Kerns nicht in Frage gestellt ist und die Spaltproduktfreisetzung deutlich unter den Auslegungsgrenzen bleibt.

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1. Problem Description

In certain reactor accidents, especially in loss-of-coolant accidents and small breaks in the primary system, the temperatures in the reactor core may reach such high values that the fuel rod cladding tubes, which are an important barrier against the escape of fission products, will lose their integrity.

It must be shown by safety analyses that, in spite of damaged fuel rods, the short term and long term coolability of the reactor core is ensured and that the fission product release from defective fuel rods does not exceed the limits specified by license criteria.

The coolability of the reactor core is influenced by

- plastic deformations (ballooning) of the Zry-cladding tubes which might result in large coolant channel blockages;

- cladding embrittlement by the uptake of oxygen and hydrogen during the accident which might result in shattering of the cladding during quenching and obstruction of cooling channels by completely oxidized cladding tube fragments; and also

- possible escape of fuel from defective or burst cladding tubes.

The extent of fission product release is influenced by:

- the temperature transient;
- the fuel pellet structure (crack formation, fragmentation); and
- the extent of cladding tube damage (i.e. the partial loss of the first barrier against fission product release into the environment).

2. Objectives

The objectives of the investigations performed in the frame of the Nuclear Safety Project (PNS) of KfK are therefore:

- to understand the relevant failure mechanisms of Zircaloy cladding tubes, and

- to provide experimentally verified methods and techniques for realistic accident analyses.

Of special importance in this context are:

- the extent of core damage, i.e.:
  - the number of ballooned and burst rods,
  - the plastic deformation and resultant coolant channel blockages,
  - the high temperature oxidation in steam and the resultant embrittlement;

- the consequences of fuel rod failures on
  - core coolability, and
  - fission product release;

- the quantification of safety margins.

3. Research Program

Fig. 1 shows a simplified overview of the research program on fuel behavior under accident conditions performed by the Nuclear Safety Project /1, 2/.

The overall program includes a large number of separate effects investigations of the material behavior, i.e. the plastic deformation of the Zircaloy cladding, the high temperature steam oxidation and embrittlement, and the interaction of the cladding material with fuel and fission products.

These separate effects tests are the basis for a computer code, which is able to describe the deformation behavior of Zircaloy cladding as a function of temperature, differential pressure, material properties and environmental impact. In addition, a qualitative understanding is attained of the relevant influences on the cladding tube behavior so that statements can be made to which extent the basic phenomena have been modeled in the code.

For the assessment of the fuel behavior code a spectrum of out-of-pile and in-pile experiments is needed which includes the system effects of single rods and bundles. In these experiments the behavior of single rods and bundles is investigated under realistic boundary conditions, i.e. under combined mechanical and chemical loads and under reactor typical cooling conditions.

Thermohydraulic separate effects tests /3, 4/ and investigations of the influence of coolant channel blockages on the coolability of deformed bundle geometries complete the program.

The theoretical models based on this experimental program are being improved step by step and integrated in the modular code system SSYST. SSYST is a development by KfK and IKE, Stuttgart; it allows a deterministic single-rod analysis over the complete LOCA transient as well as a probabilistic analysis of the extent of damage in the reactor core.

Fig. 1 does not show the investigations of fission product release which were performed in close cooperation between the Nuclear Research Center Karlsruhe and the Oak Ridge National Laboratory. In Oak Ridge these investigations concentrated on loss-of-coolant accidents, in Karlsruhe on core meltdown accidents.

Most of the research work on fuel behavior was started in 1973 and concentrated on loss-of-coolant accidents with double ended cold leg breaks.

The typical boundary conditions for these accidents are a rapid decrease of the system pressure, temperatures up to 1200 °C and times of a few minutes. (Fig. 2)

Since 1981, as a result of American and German risk studies the investigations are being extended more and more also to other transients, especially small breaks and so-called anticipated transients with and without scram.

This paper will describe the status and the most important results of the PNS fuel behavior investigations under LOCA conditions and will give a short review of the future activities.

4. Results

4.1 The Deformation and Burst Behavior of Zircaloy Claddings and the Coolability of Deformed Core Geometries under LOCA Conditions

The deformation and burst behavior of Zircaloy cladding tubes is a very complex problem and influenced by a lot of separate effects all of them greatly dependant on the temperature, the time and the heatup rate, respectively.

In the temperature range of 500 to 800 °C deformation is governed primarily by a dynamic equilibrium between hardening and annealing processes. In the range of 820 to 980 °C a phase transformation of the hexagonal closely packed α-phase into the body centered cubic β-phase takes place.

In parallel, also chemical reactions occur like oxidation of the cladding tubes in steam and interaction between the cladding and the fission products which may also influence severely the ductility and strain of the claddings.

Due to extensive investigations conducted in Germany, USA, Japan, France, the United Kingdom and other countries all these effects are presently well understood and can be described in quantitative terms for given pressure and temperature transients /5,6,7,8,9/.

Fig. 3 shows the burst temperature of Zircaloy cladding tubes as a function of the burst pressure. The figure shows that for a given pressure the burst temperature is influenced by the heatup rate: With increasing heatup rate the burst temperature increases /9/.

It is evident that the in-pile data of the FR-2 tests fit well in the scatter band of the out-of-pile results /10, 11/.

The burst temperature is an important criterion for the number of defective rods and the estimate of the resultant fission product release. At least for the α-phase of Zircaloy, the burst temperature can be predicted with a high degree of precision by available models.

The burst strain is influenced by the heatup rate and the azimuthal and axial temperature distributions on the cladding:

Fig. 4 shows the burst strain as a function of the azimuthal temperature difference on the cladding. The data were measured in out-of-pile experiments with indirectly heated electrical fuel rod simulators. It can be seen that for large azimuthal temperature differences cladding deformation is concentrated on the hot spot, where also the break occurs. For small azimuthal temperature differences the deformation of the cladding is distributed uniformly on the circumference and the resultant strains are relatively large /14/.

Fig. 5 shows the combined effects of heatup rate and azimuthal temperature distribution: In the high α-region around 800 $^{\circ}C$ which is important for loss-of-coolant accidents, small heatup rates and uniform temperature distributions on the circumference of the cladding result in high strains. Especially creep rupture tests with direct electrically heated cladding tubes (dotted line) result in extremely uniform temperature distributions and hence in high strains /12/.
High heatup rates and nonuniform temperatures on the cladding circumference result in small burst strains.

In the β-region the heatup rate shows the opposite effect, due the increasing oxygen embrittlement of the cladding.

Out-of-pile bundle tests, especially those performed under the REBEKA program of KfK, involving fuel rod simulators of high simulation quality and including thermohydraulic boundary conditions result in very nonuniform temperature distributions and hence in relatively small burst strains /13, 14/.

The figure shows that most of the rods have undergone circumferential strain of less than 60 % if the tests were performed under realistic boundary conditions. In-pile experiments performed in the FR2-reactor of KfK with fresh and highly burnt-up fuel rods (maximum burnup: 36 000 MWd/t) do not differ very much from the results of these out-of-pile experiments (see. Fig. 6) /10, 11/.

To determine the coolability of the core, however, not only the circumferential strains of single rods but also the axial distribution of ballooned areas in the core must be known:

Fig. 7 shows the circumferential strain of the nine central fuel rod simulators and the resultant coolant channel blockage from REBEKA bundle test 3 /14/. This test was performed with a 25-rod bundle of full length under thermohydraulic boundary conditions which are typical for the refill and reflood phases of a loss-of-coolant accident. As a consequence of the axial power distribution and due to the influence of the two-phase flow which is not in equilibrium and which is disturbed at each spacer grid, only relatively small and local cladding deformations occur. The maximum coolant channel blockage is only about 52 % (see also Fig. 8, 9).

This result is confirmed by the REBEKA-5 test, which was recently performed using a bundle of 7 x 7 rod simulators of full length and a cosine shaped power profile. Also in this test the maximum coolant channel blockage was only 52 % (Fig. 10, 11).

Separate effects tests within the FEBA program on the influence of coolant channel blockages on the effectiveness of emergency core cooling indicate that even coplanar blockages up to 90 % are coolable and the maximum cladding temperature in the reflood phase is not much higher than the temperature of unblocked fuel elements (Fig. 12), /15, 16/.

It should be stated, however, that these investigations were performed on a bundle containing not more than 25 fuel rods and with forced reflood. So it is necessary to confirm the results by tests relating to larger bundle configurations with partial blockages in which the bypass flow around the blockage can be simulated in a more representative manner. Such experiments are planned within the frame of the international 2D/3D-project /30/.

Tests on the long-term coolability of the flooded core with blocked arrays have shown that coolant channel blockages up to 90 % did not jeopardize the coolability of the core as long as it was possible to cover the core with water (Fig. 13) /17/.

In general, it can be stated that in case of nonuniform temperature distributions which are nearly always ensured under loss-of-coolant accident conditions the maximum cladding tube strains are concentrated on relatively small local areas of the cladding circumference and that large circumferential and axially extended ballooning which may question the coolability of the core are not to be expected.

4.2 High Temperature Steam Oxidation and Embrittlement of the Cladding Material

As a result of extensive investigations performed by KfK, ORNL and JAERI the kinetics of high temperature steam oxidation are fully understood today and can be discribed with a good accuracy by computer codes such as SIMTRAN or BILD 5.

These codes are capable of calculating the oxygen uptake, the thicknesses of the ZrO_2 layer and the oxygen stabilized α-phase of the Zircaloy as well as the oxygen concentration profile under isothermal and transient conditions /18, 19/.

Fig. 14 shows the increase of the ZrO_2 layer as a function of time at different temperatures. The agreement between measured and calculated data is very good.

Fig. 15 shows the oxygen uptake as a function of time at different temperatures. The oxygen uptake during LOCA typical times can be described by cubic and parabolic time laws, respectively. Especially at temperatures above 1000 ^{o}C it is about 25 % less than the Baker-Just correlation would predict.

Under loss-of-coolant accident conditions this means less oxidation, less hydrogen production and less exothermal energy than assumed in safety analyses.

The present acceptance criteria for emergency core cooling systems are to ensure by limitation of the peak cladding temperature to 1200 ^{o}C and the calculated oxidation to 17 % of the cladding tube wall thickness, that embrittled cladding tubes resist also the thermoshock stresses during quenching and the mechanical stresses during withdrawal and handling of the fuel elements.

All investigations performed up to now have shown that under LOCA conditions the 17 %-criterion in combination with the Baker-Just correlation is indeed very conservative.

Chung and Kassner (ANL) found that it is more adequate to define a minimum remaining wall thickness of the ductile β-phase of the Zircaloy cladding, rather than a maximum temperature and oxidative penetration /31/.Their recommendation is (Fig. 16):

- for cladding to have the capability to withstand thermal shock during reflood the calculated thickness of the cladding with $\leq$ 0,9 w/o oxygen based on the average wall thickness at any axial location shall be greater than 0,1 mm;

- for cladding to have the capability to withstand fuel handling, transport and storage the calculated thickness of the cladding with $\leq$ 0,7 w/o oxygen based on the average wall thickness at any axial location shall be greater than 0,3 mm.

These criteria have the advantage to be applicable irrespective of the oxidation temperature, initial wall thickness and deformation. However, they need a good code to calculate the oxygen concentration in the cladding.

4.3 Influence of Fuel and Fission Products on the Deformation and Burst Behavior of Zircaloy Cladding Tubes

Out-of-pile investigations of the influence of fission products on the deformation behavior of Zircaloy cladding have shown that iodine has the potential to reduce drastically the burst strains and the tangential burst stresses due to stress corrosion cracking.

The critical iodine concentration which is needed to induce stress corrosion cracking is, however, strongly dependent on the temperature (Fig. 17, 18).

These figures shows that at temperatures > 700 ^{o}C the critical iodine concentration is higher than the iodine inventory in high burnup fuel rods.

This means that under loss-of-coolant accident conditions when the burst temperatures are generally > $700^{o}C$, the influence of iodine can be neglected /22, 23/.

Under different accident conditions, especially during so-called anticipated transients, the iodine influence must be taken into account.

Chemical reactions between the Zircaloy cladding and the fuel occur at a contact pressure which can be expected for instance under power cooling mismatch conditions.

At high temperatures the UO_2 will be partially reduced by the Zircaloy-4 to form oxygen-stabilized α-Zr(O) and uranium metal (Fig. 19). The uranium reacts with oxygen-poor Zircaloy to form a (U, Zr) alloy which lies between two α-Zr(O) layers. The sequence of the UO_2/Zry reactions zones is always the same as shown in figure 19 and below:

$$UO_2 \rightarrow (\alpha\text{-}Zr(O)_a + (U,Zr)) \rightarrow (U,Zr)\ \text{alloy} \rightarrow \alpha\text{-}Zr(O)_b \rightarrow \text{Zry-4}.$$

The Zircaloy cladding can oxidize nearly as rapidly from reaction with UO_2 (inner corrosion) as from reaction with steam at the outside surface (high-temperature steam corrosion) /24/; Fig. 20.

4.4 Fission Product Release in a Loss-of-Coolant Accident

The previous studies of fission product release mainly relate to loss-of-coolant accidents and hypothetical core meltdown accidents. The results have been derived from out-of-pile experiments performed at ORNL and KfK, respectively. To a limited extent, additional information can be obtained from some power cooling mismatch experiments performed in the Power Burst Facility and from hot cells annealing experiments made at KfK within the frame of post-irradiation examinations following the in-pile experiments in the FR2 reactor.

Especially the experiments performed at ORNL with induction heatet fuel rod specimens from fuel elements burnt-up in power reactors support the previous results that up to maximum fuel rod temperatures of $1200^{o}C$, i.e., in successfully managed loss-of-coolant accidents, the measured release of iodine and cesium is much smaller than the gap release described in the US and German Risk Studies.

In Fig. 21 the gap inventory, the gap release and the total release of xenon, krypton, cesium and iodine are listed following the assumptions made in the Risk Studies.
These values are compared with calculations of ORNL using a model which had been based on their experiments.

The Oak Ridge results indicate that especially the total release of cesium and iodine is smaller by a factor of 200 and 60, respectively, than assumed in the German and American Risk Studies. These Oak Ridge data are consistent with the results of annealing experiments on fission gas release performed at KfK.
The assumptions contained in the Risk Studies are evidently very conservative.

In-pile experiments in the HFR reactor in Petten are planned by KFA Jülich to investigate the gap release of iodine from defective fuel rods. These investigations are to confirm the out-of-pile results. They will concentrate on the following problems:

- The amount of iodine release from defective fuel rods,
- the chemical form of the released gas,
- the influence of the surrounding medium (water, steam),
- the influence of pressure, temperature and time,
- the influence of nuclear parameters (burnup).

It is expected that the first results of this program will be available in early 1984.

4.5 Code System SSYST for the Description of Fuel Rod Behavior under LOCA Conditions

The code system SSYST was developed by KfK and IKE Stuttgart for the description of the transient fuel behavior and for realistic estimates of the scope of damage occurring in a reactor core under accident conditions /25, 26/.

Fig. 22 shows, in a very simplified form, the structure of the program system.

SSYST has a modular structure and consists of a central data base, a control program and different separate moduls. Before starting the transient fuel rod analysis it is necessary to define the rod initial conditions and the thermohydraulic boundary conditions by means of special codes like COMETHE or RELAP.

The transient fuel rod behavior, especially the deformation as a function of temperature, pressure differential and environmental impact is then calculated by a number of separate modules which describe the relevant physical and chemical phenomena.

SSYST, like other fuel behavior codes available, is a single-rod model which is being systematically assessed and improved.

In parallel to these activities, different methods have been developed for estimating the extent of damage in a reactor core under loss-of-coolant accidents.

In most of the safety analyses performed up to now so-called deterministic conservative methods have been used. In these calculations the behavior of different groups of rods with similar power and burnup history was investigated using very pessimistic assumptions, and the overall damage in the core and the resultant coolant channel blockage were estimated on that basis.

This procedure can lead very easily to very unrealistic and pessimistic estimates regarding the deformation of the rods and the resultant coolant channel blockage and fission product release.

More realistic estimates can be made by means of probabilistic methods as developed by IRE and INR of KfK. These methods were elaborated to investigate the influence of uncertainties in the model parameters and input variables on the interesting system answers, i.e. cladding temperature and strains. A combination of the so-called response surface method and Monte Carlo techniques was used (Fig. 23). These developments have been completed and can be used for analysing single rods as well as for the analysis of fuel rod bundles /27, 28, 29/.

Fig. 24 shows as an example the results of a probabilistic damage analysis in a fuel rod bundle.

In the upper part of the figure, the probability density function of the maximum cladding temperature of a fuel rod is shown.

The lower part of the figure shows the probability density function of the circumferential strain.

It must be underlined that the methods of probabilistic damage analyses must be assessed and, without any doubt, improved. However, they are already a valuable tool in performing sensitivity studies and in identifying the dominant parameters which influence the system answer. In addition, they allow conclusions on the safety margins between tolerable system answers and critical thresholds.

5. Summary and Conclusions

The investigations into the fuel behavior under reactor accident conditions, similar to the other studies on emergency core cooling, have initially been concentrated on loss-of-coolant accidents with double ended cold and hot leg breaks in the primary system.

Most of these investigations have been terminated and have yielded important results. Especially the relevant phenomena, such as the deformation and failure of the cladding, the high-temperature steam oxidation, the interaction of the cladding with fuel and fission products, as well as the impact of thermohydraulics on the deformation behavior of the cladding are well understood today.

Deformation models and failure criteria are available. They are being continuously verified and improved the same as the whole program system SSYST.

Single-rod and bundle experiments which were performed under realistic boundary conditions indicate that the deformation of the fuel rods and the resultant cooling channel blockage do not question the short-term and long-term coolability of the core as long as emergency core cooling systems work correctly.

Additional investigations of the fuel behavior under LOCA conditions at KfK concentrate on out-of-pile bundle experiments aimed at providing knowledge of rod-to-rod interaction and potential failure propagation. Most of this work will be finished in 1983.

It is expected that the present understanding of fuel behavior under LOCA conditions will be confirmed by in-pile experiments in PHEBUS, NRU and SUPER SARA which will be available in the near future.

Of special importance for the future work on fuel behavior is a shifting of emphasis from loss-of-coolant accidents with large rupture cross sections to small breaks and special transients which, in combination with failing safety systems, may lead to severe core damage. The necessity to investigate these transients in more detail became already evident some years ago as a result of the American and German Risk Studies. It was underlined by the Three Miles Island Accident.

These investigations on severe core damage which are already on the way or at the planning stage concentrate on fuel behavior in relatively slow transients starting under normal operation conditions and ending at the melting temperature of Zircaloy.

Of special importance are:

- The chemical interaction between Zircaloy and fuel and the form, the extent and effects of liquified phases in the gap between Zircaloy cladding and UO_2 pellets;

- the oxidation kinetics and the embrittlement of the Zircaloy and the structure material at temperatures > 1200 ^{o}C, longer holding times and possible steam starvation;

- the resultant hydrogen production at high temperatures;

- the coolability of a severely damaged reactor core; and

- the fission product release during slow transients and temperatures > 1200 ^{o}C.

On the basis of out-of-pile experiments performed by KfK the most relevant phenomena are already qualitatively known.

The future activities will contribute to quantify these phenomena and to incorporate the resultant computer models into the program system SSYST.

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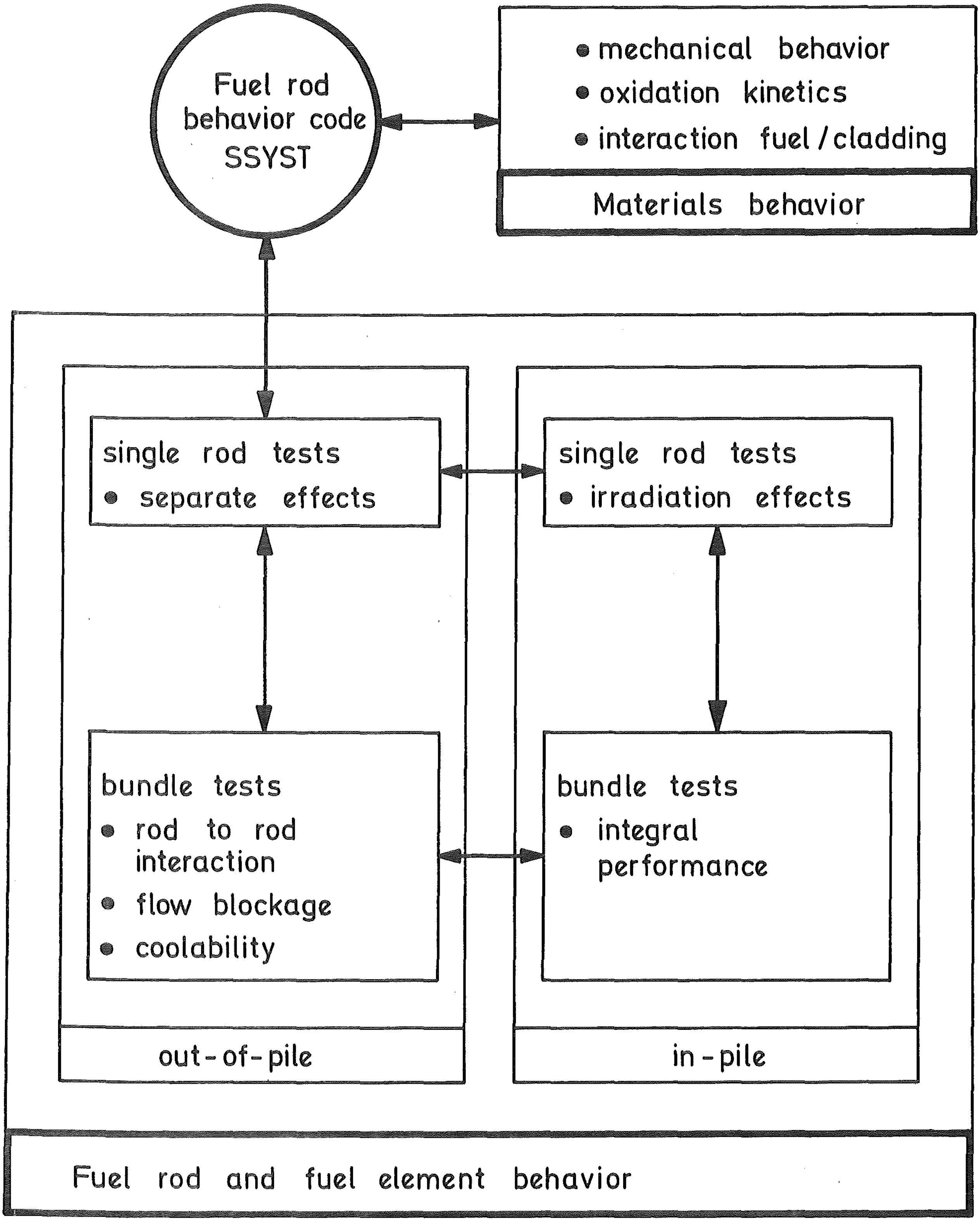


Fig. 1

General approach of the LWR fuel rod behavior research

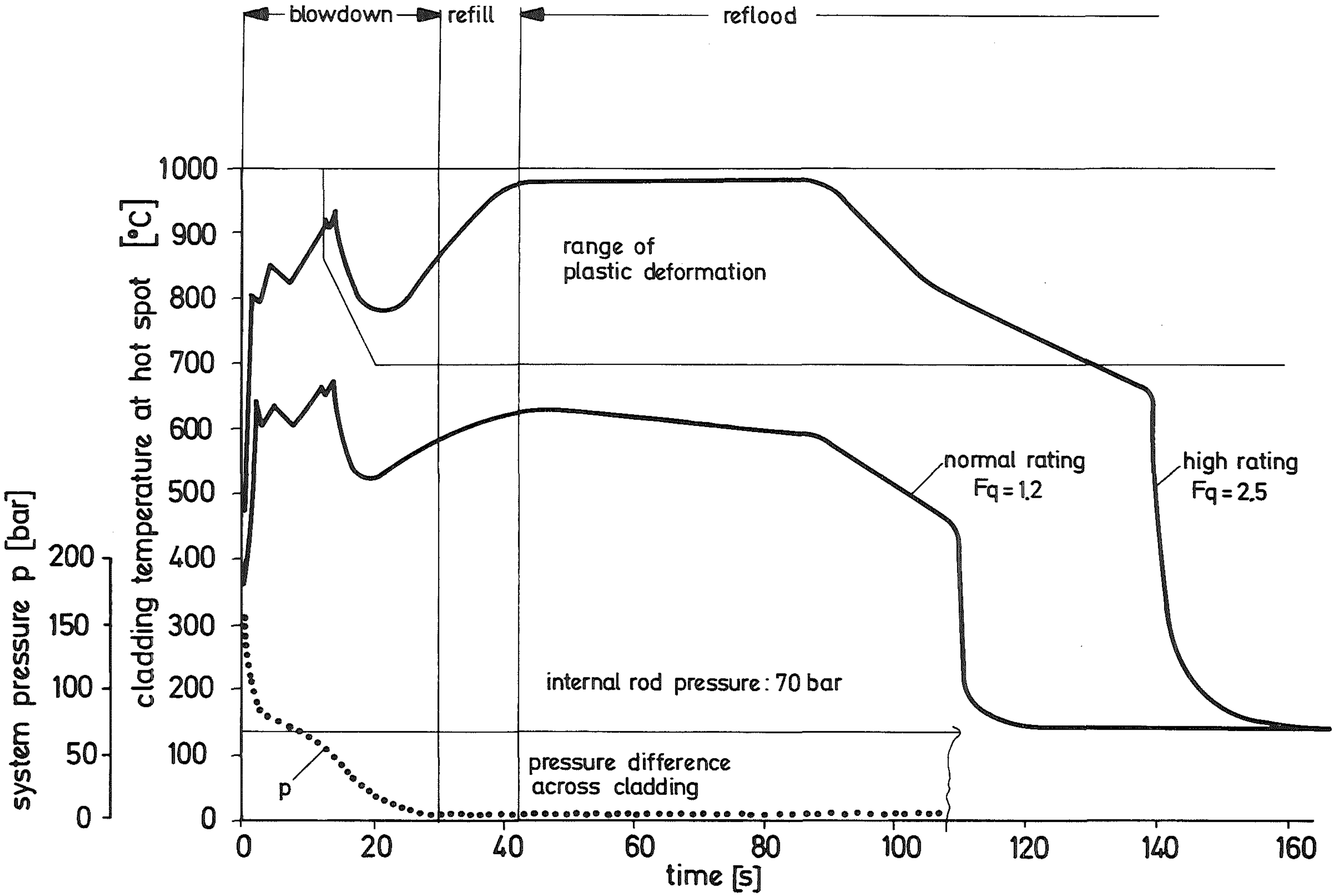
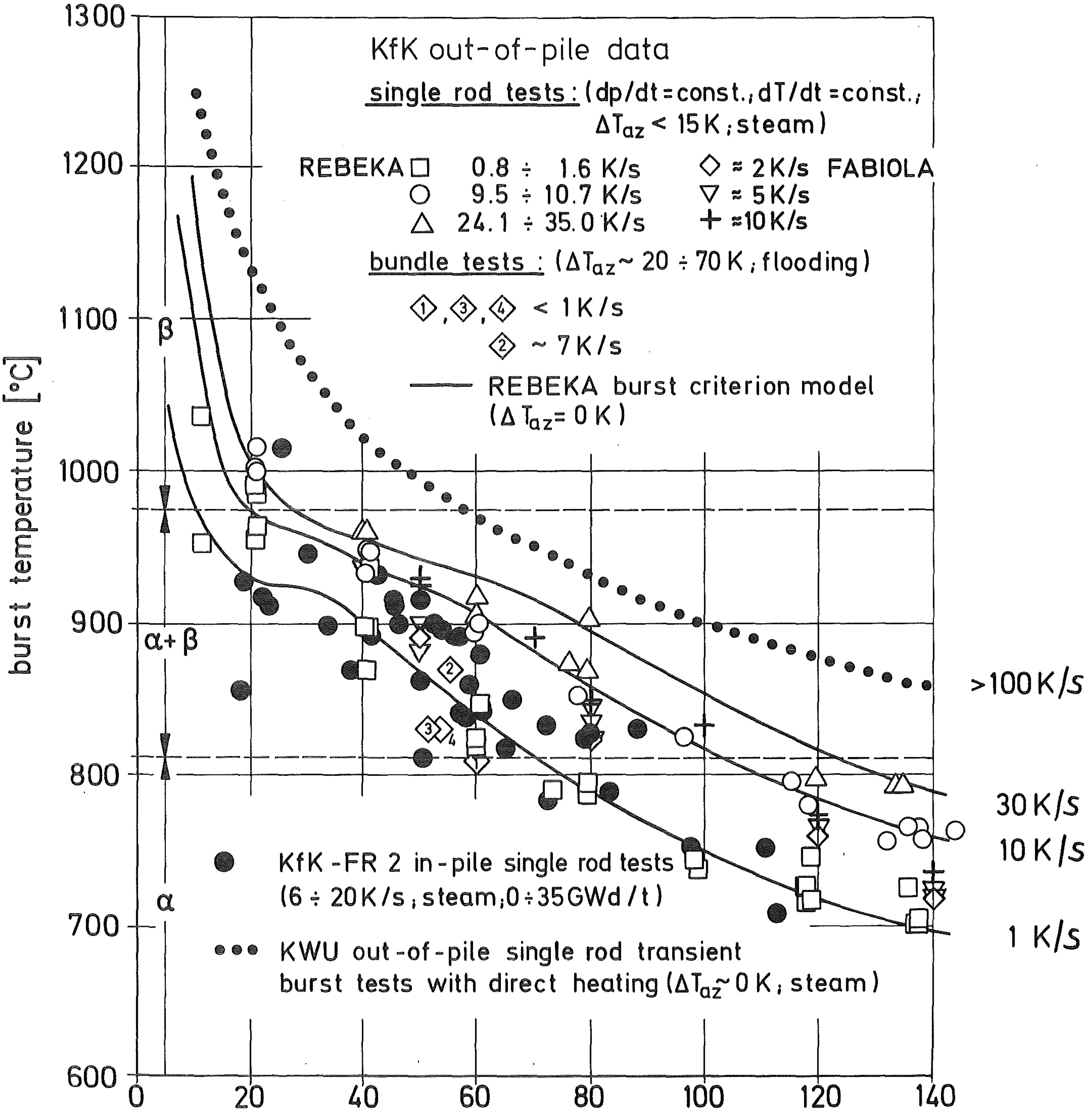


Fig. 2

Fuel rod cladding loading in a 2F-cold leg break LOCA



KfK

Fig. 3

Burst temperature vs. burst pressure of Zircaloy claddings

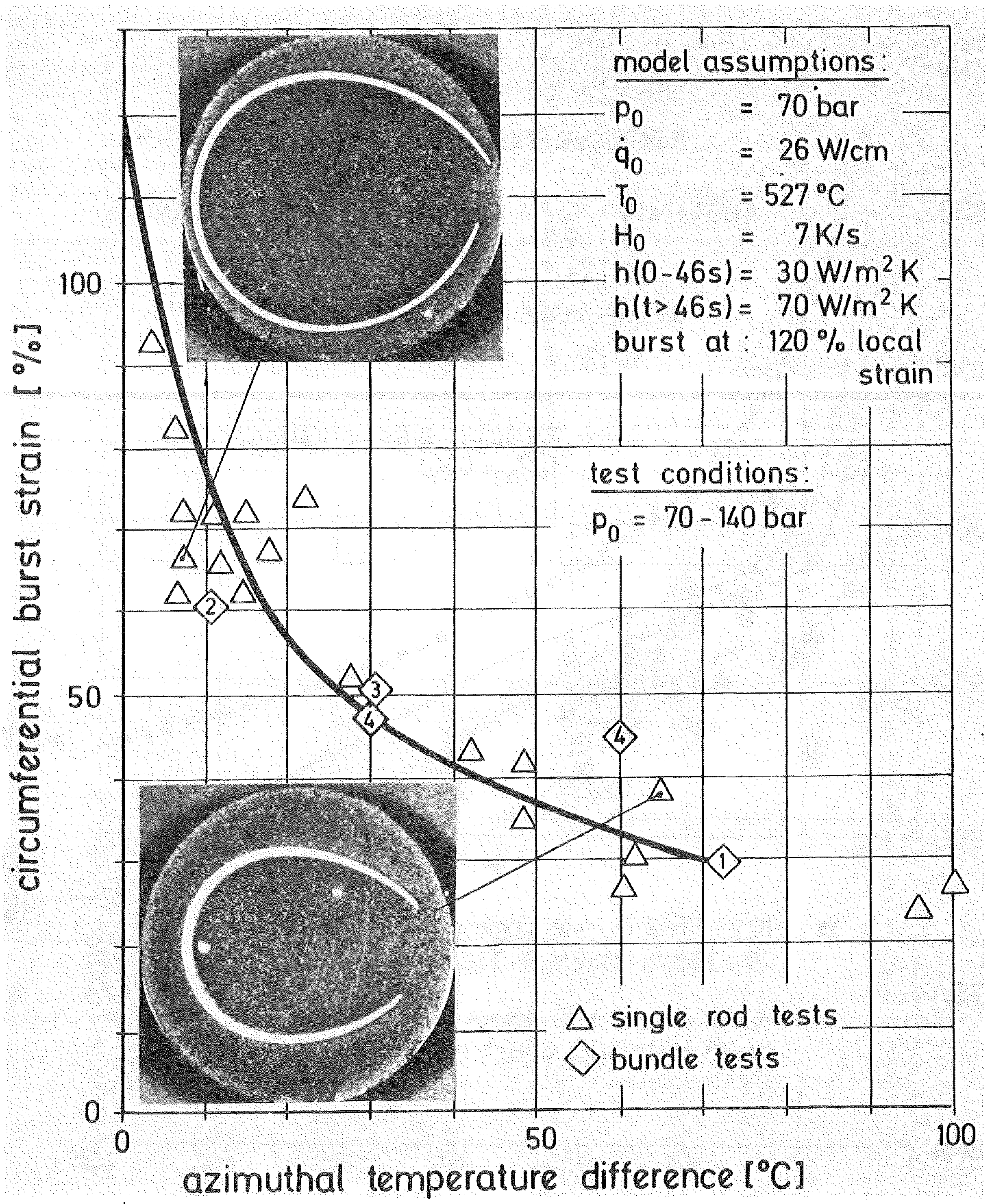


Fig. 4

Burst strain vs. temperature difference

(SSYST / AZI-prediction and test results)

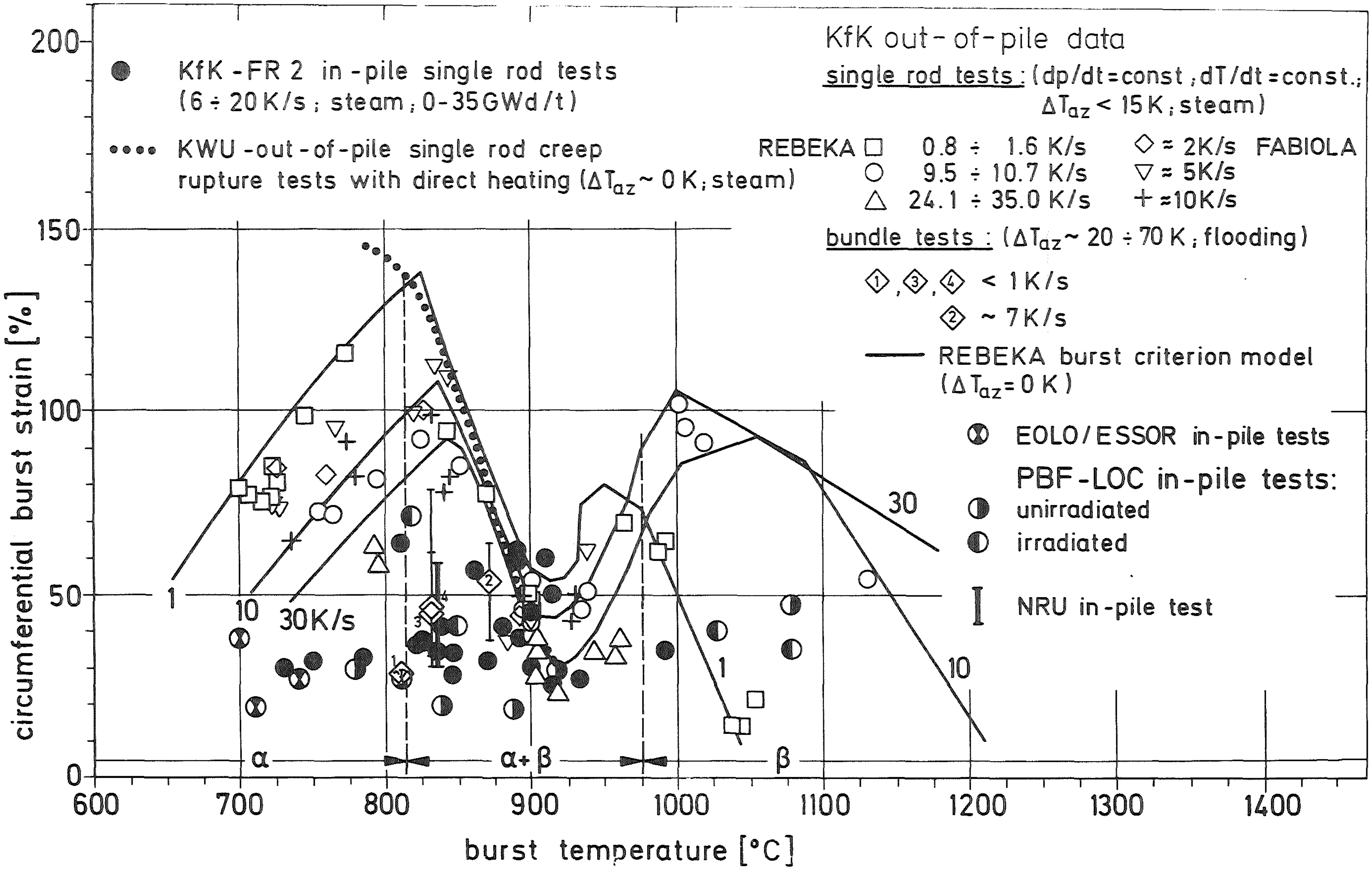


Fig. 5

Burst strain vs. burst temperature of Zircaloy claddings

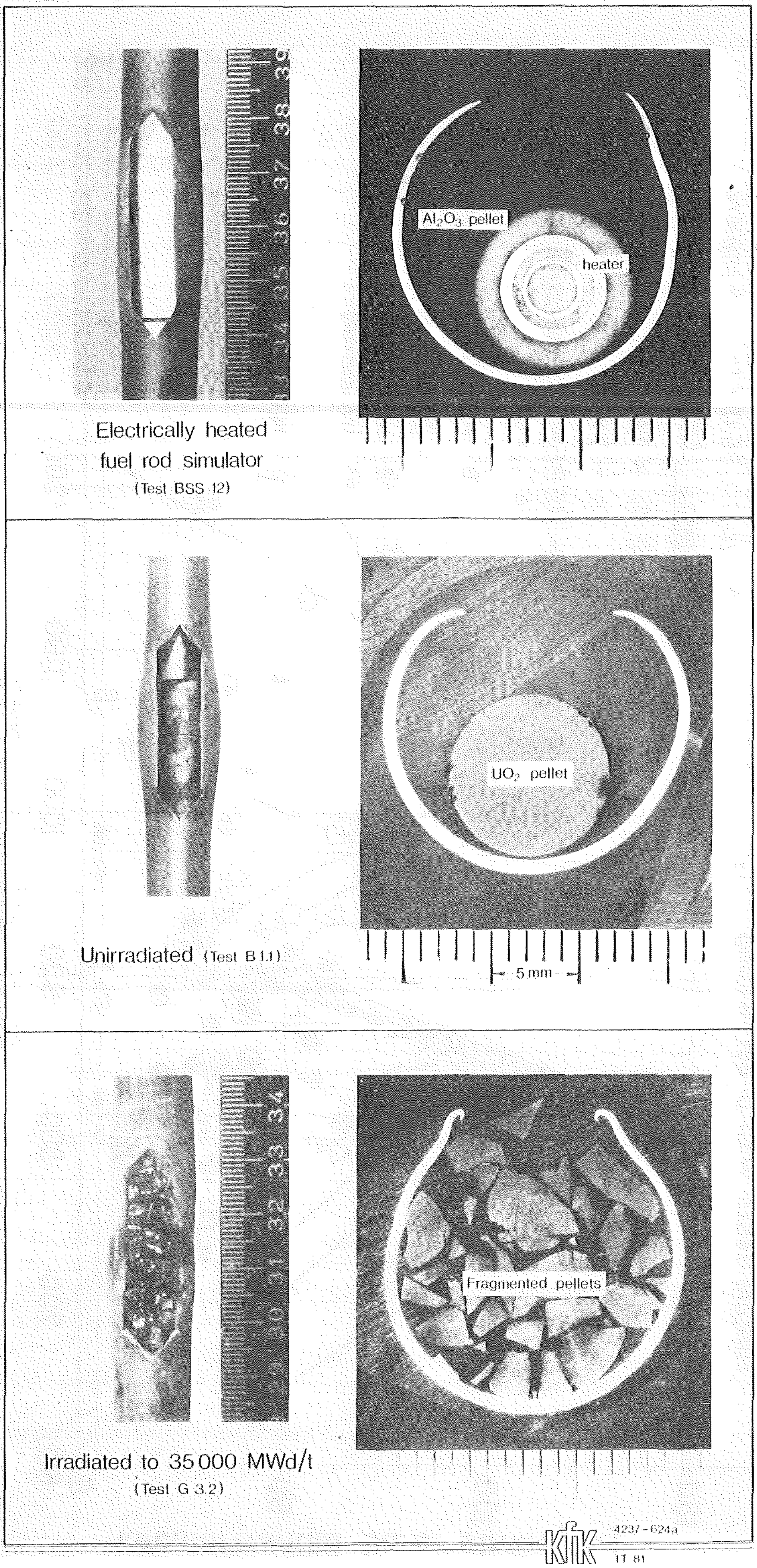


Fig. 6 **FR 2 In-pile Tests with fuel rods and rod simulators.**
Views and cross sections of rupture regions.

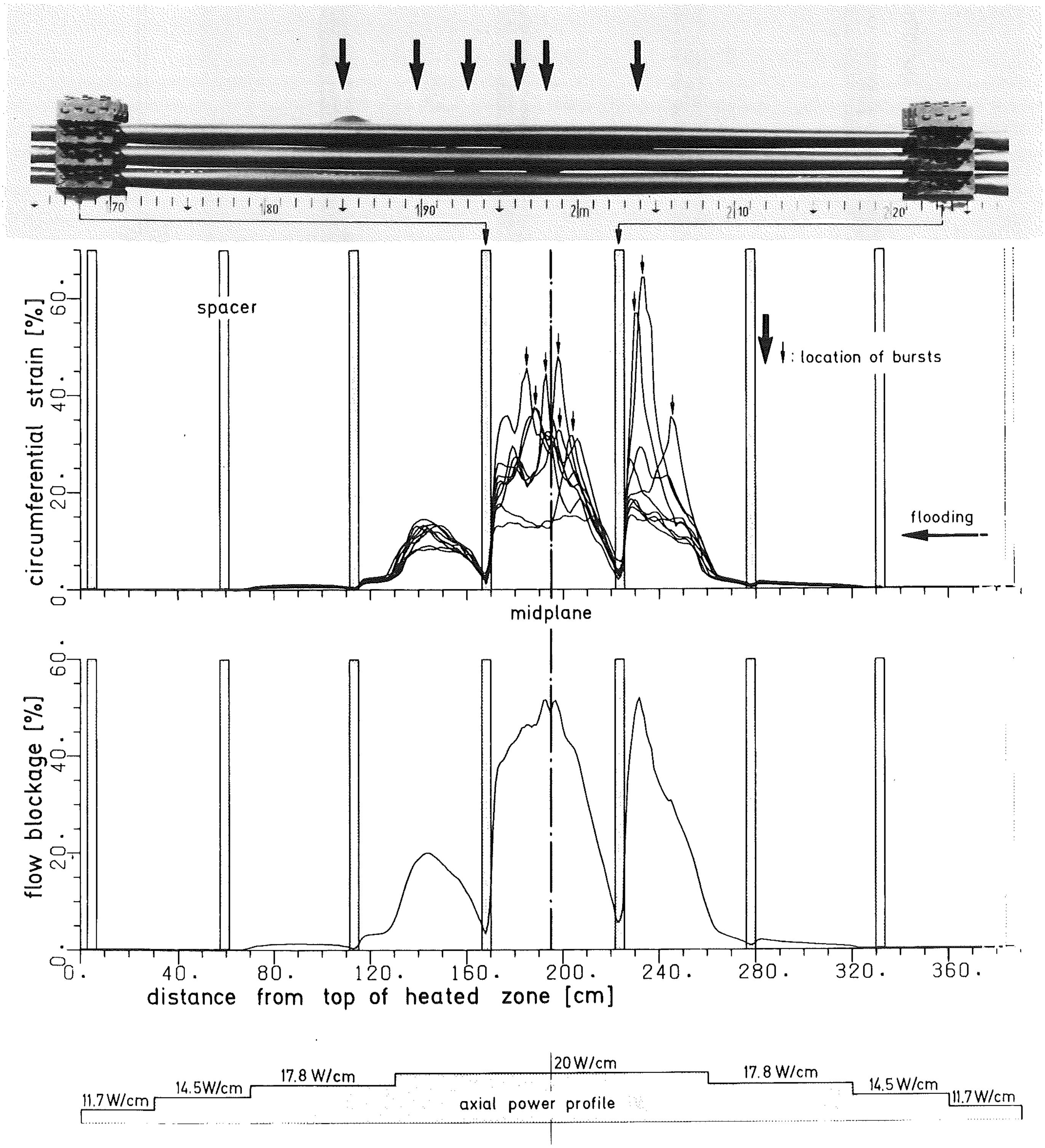


Fig. 7

REBEKA 3

circumferential strain of the 9 Zircaloy claddings and flow blockage

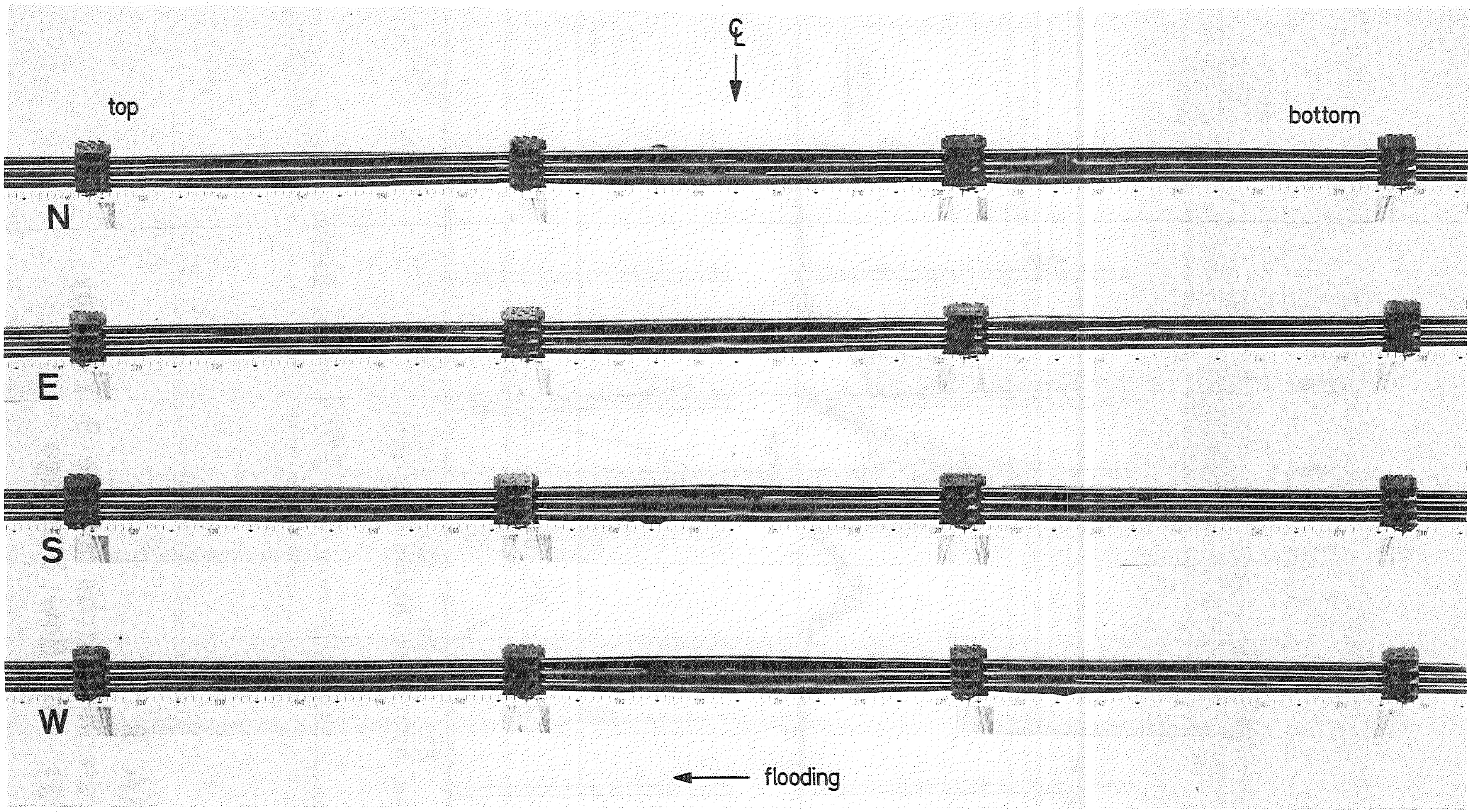
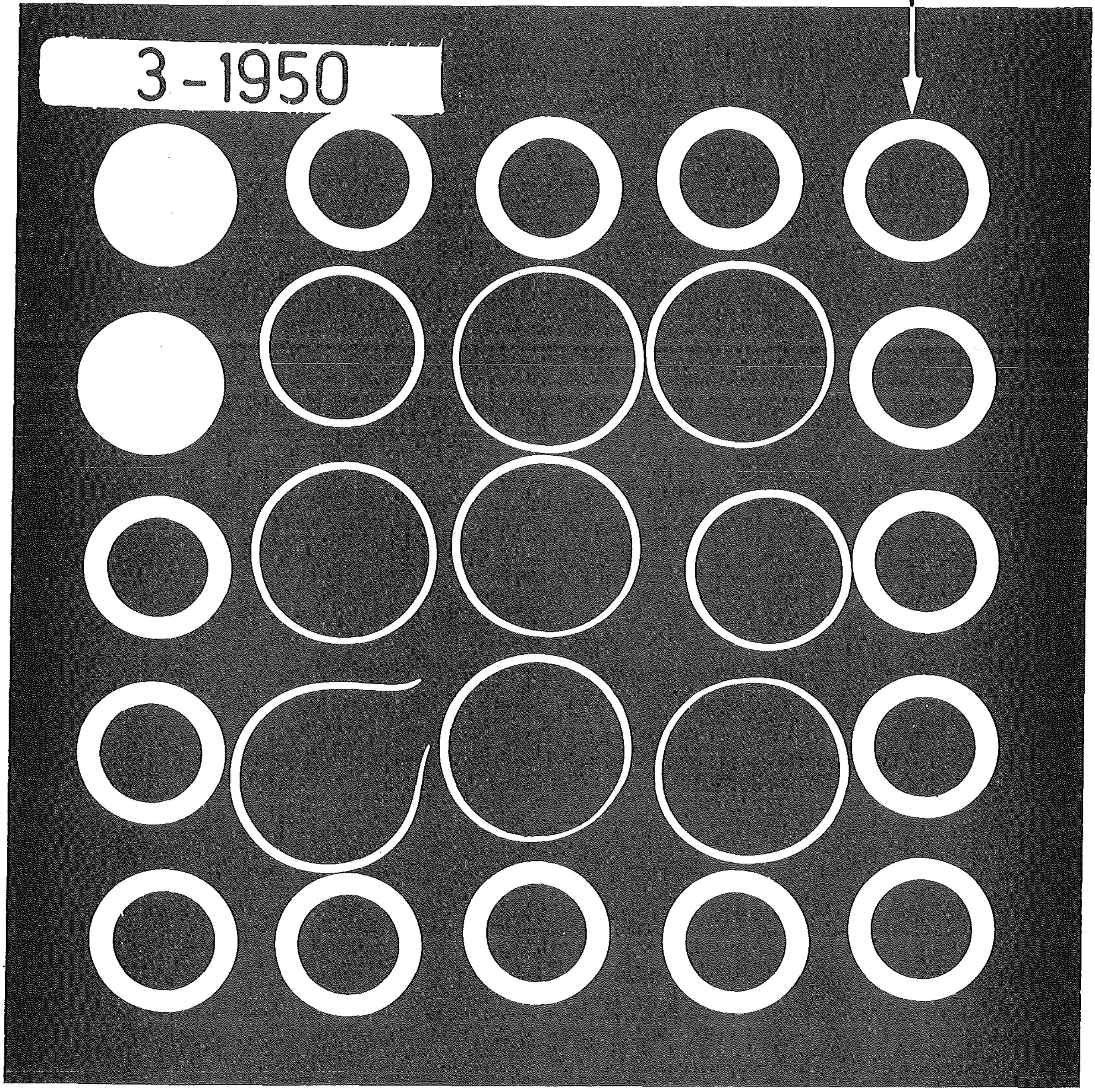


Fig. 8

REBEKA 3

deformation between the interior grid spacers, guard fuel rod simulators removed dummy tubes



axial midplane

max. flow blockage : 52 %

Fig. 9

REBEKA 3

bundle cross section

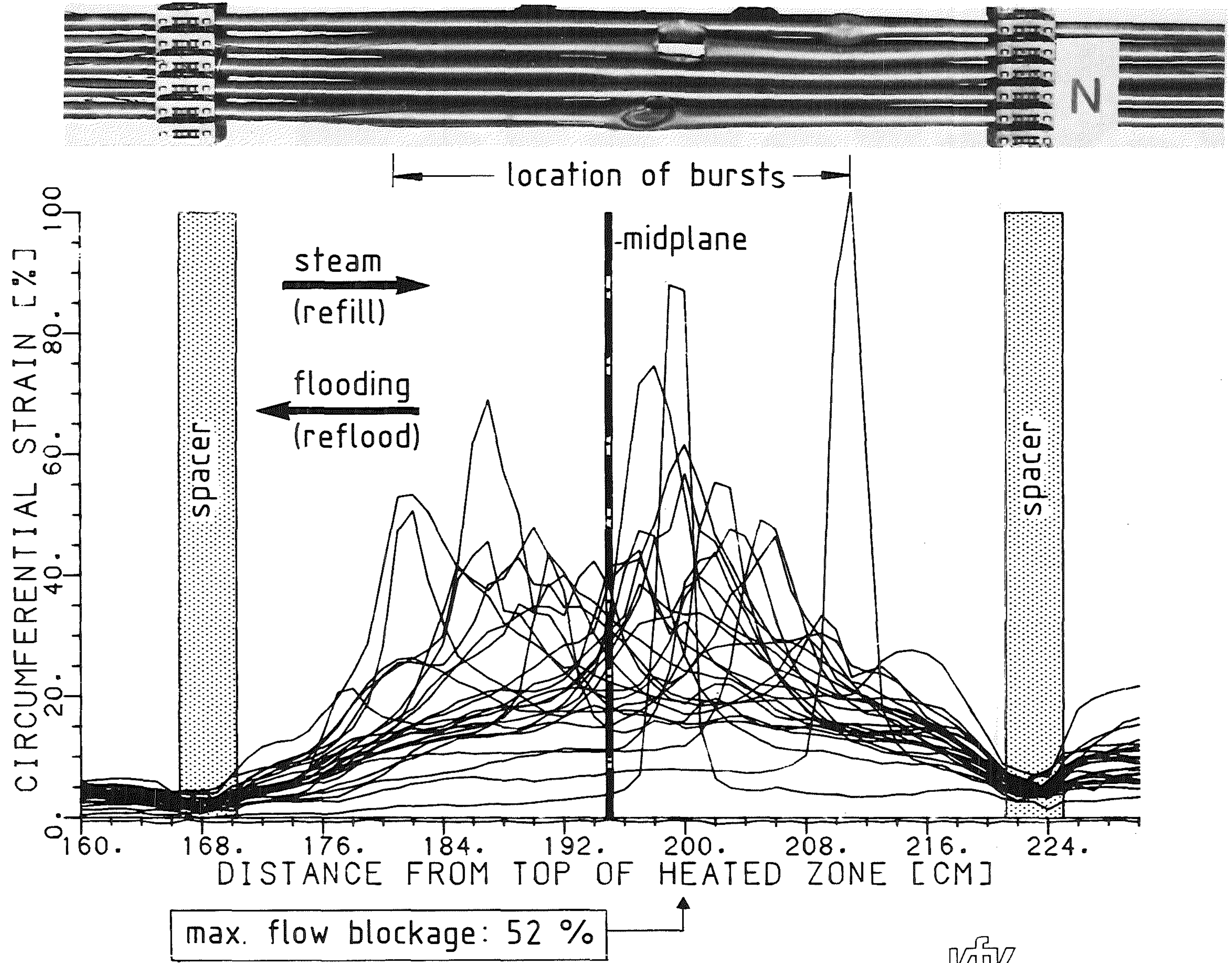
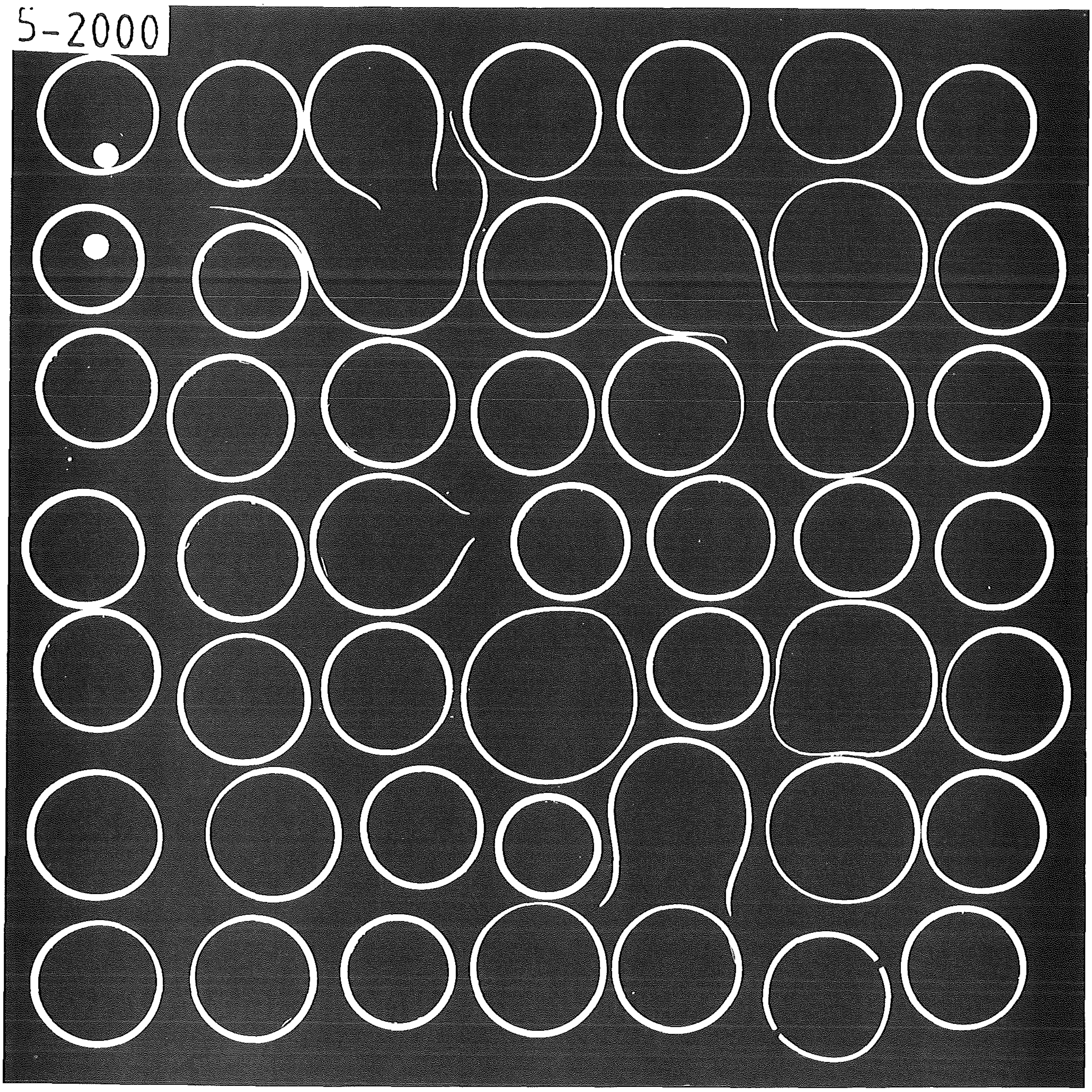


Fig. 10

REBEKA 5

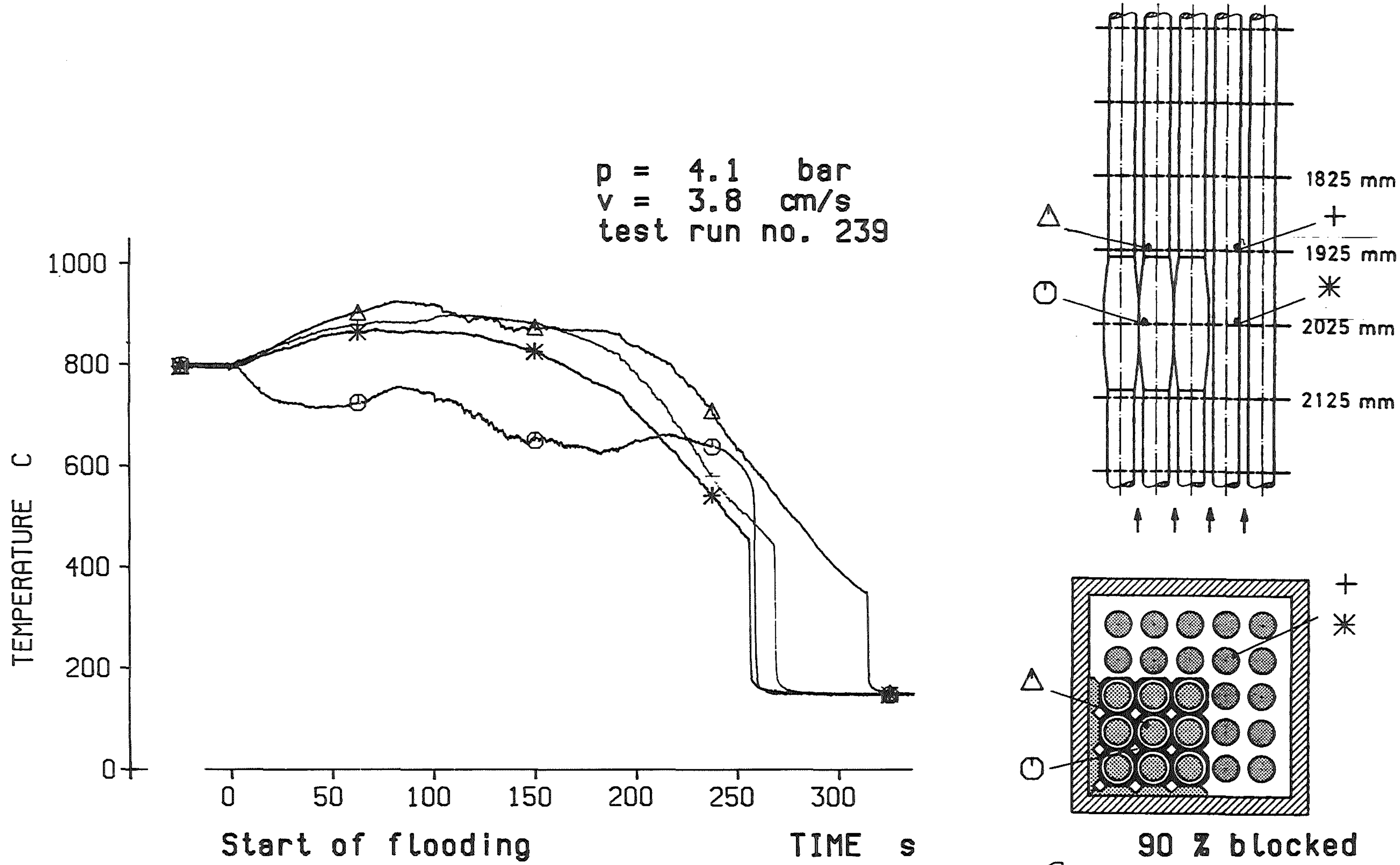
axial deformation profile of the inner 25 rods axial plane: 2000 mm from top.



KfK

Fig. 11

REBEKA 5 bundle cross-section
axial level of max. flow blockage



FEBA KfK IRB

Fig. 12

CLADDING TEMPERATURES IN PARTLY BLOCKED 25-RØD BUNDLE

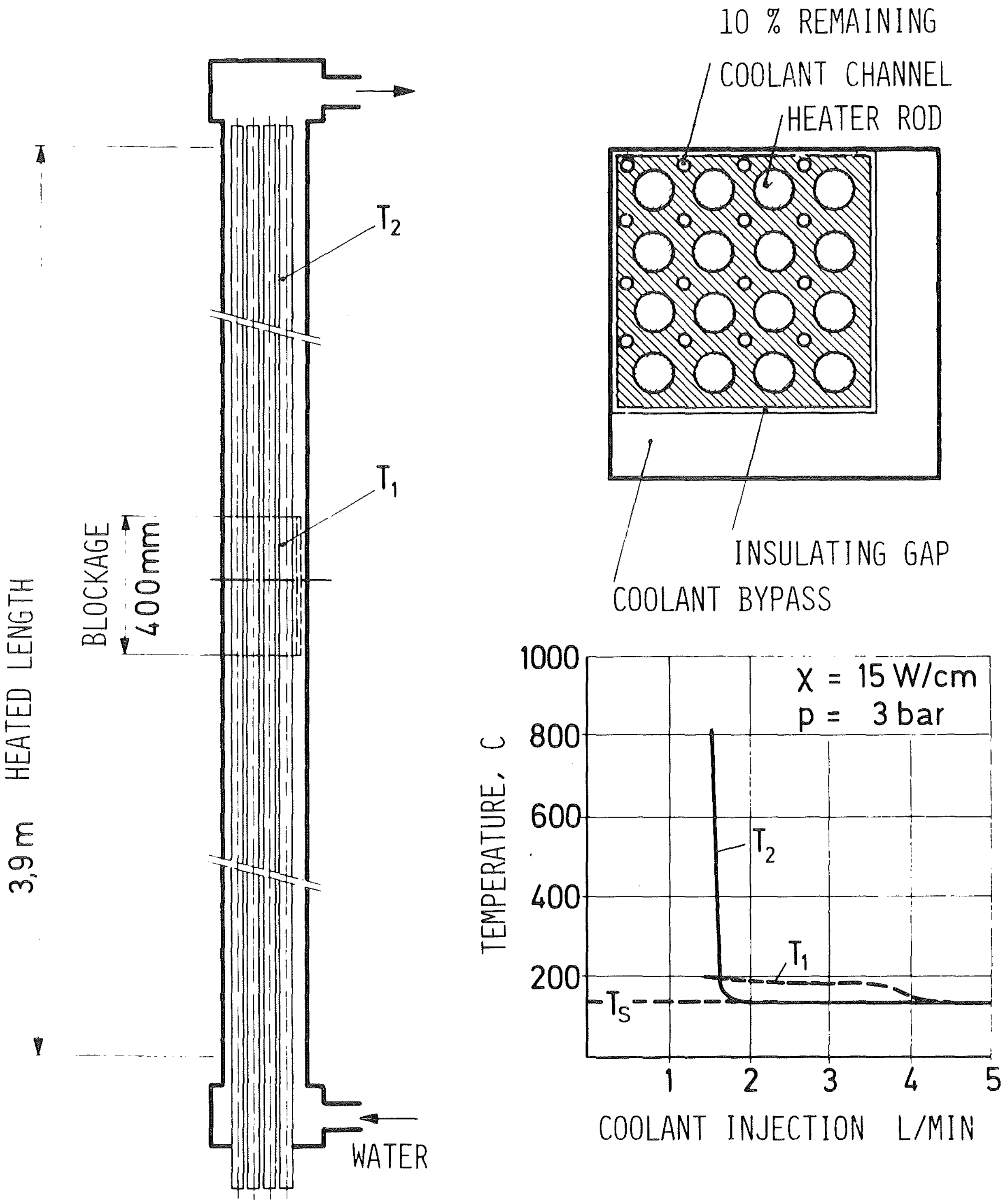
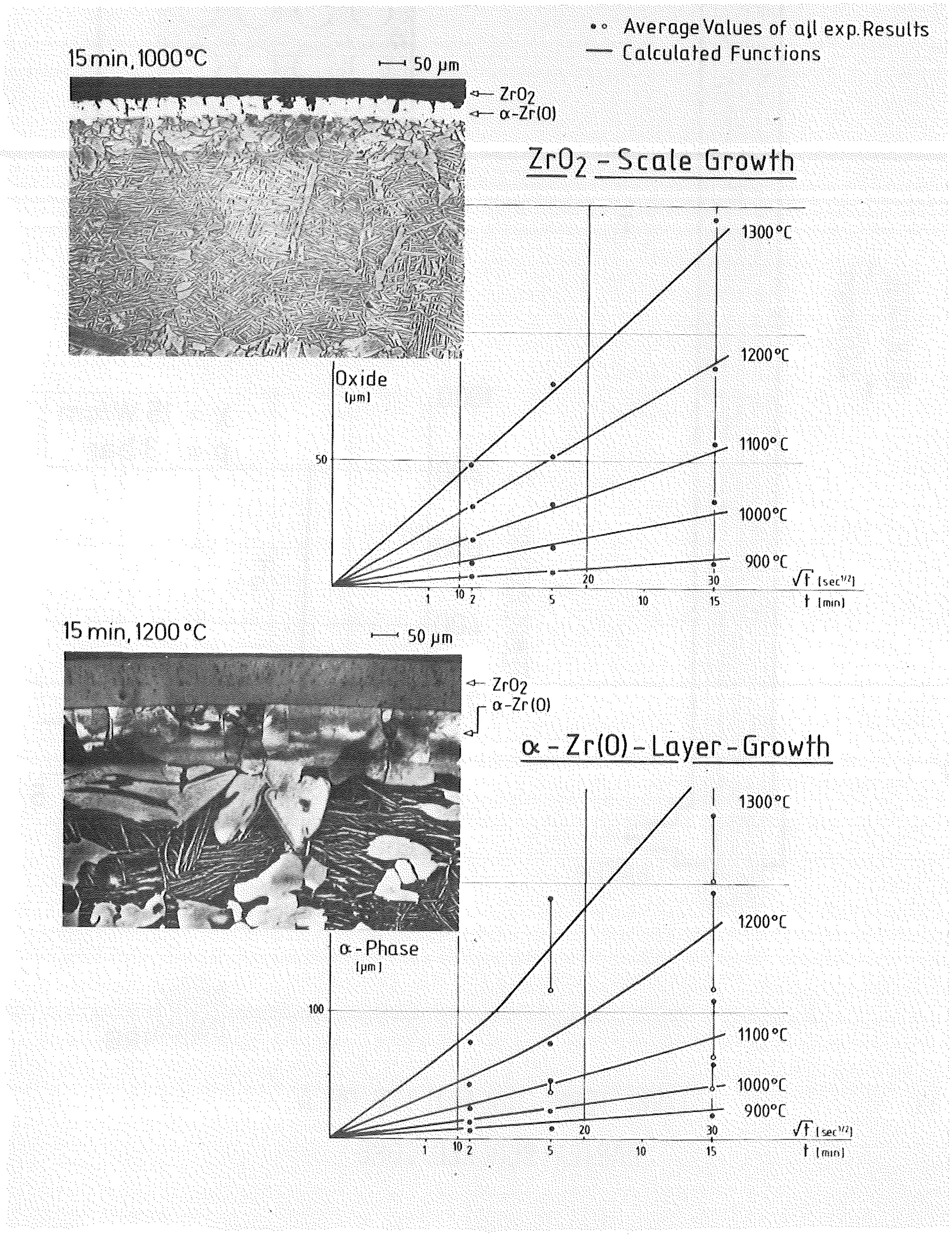
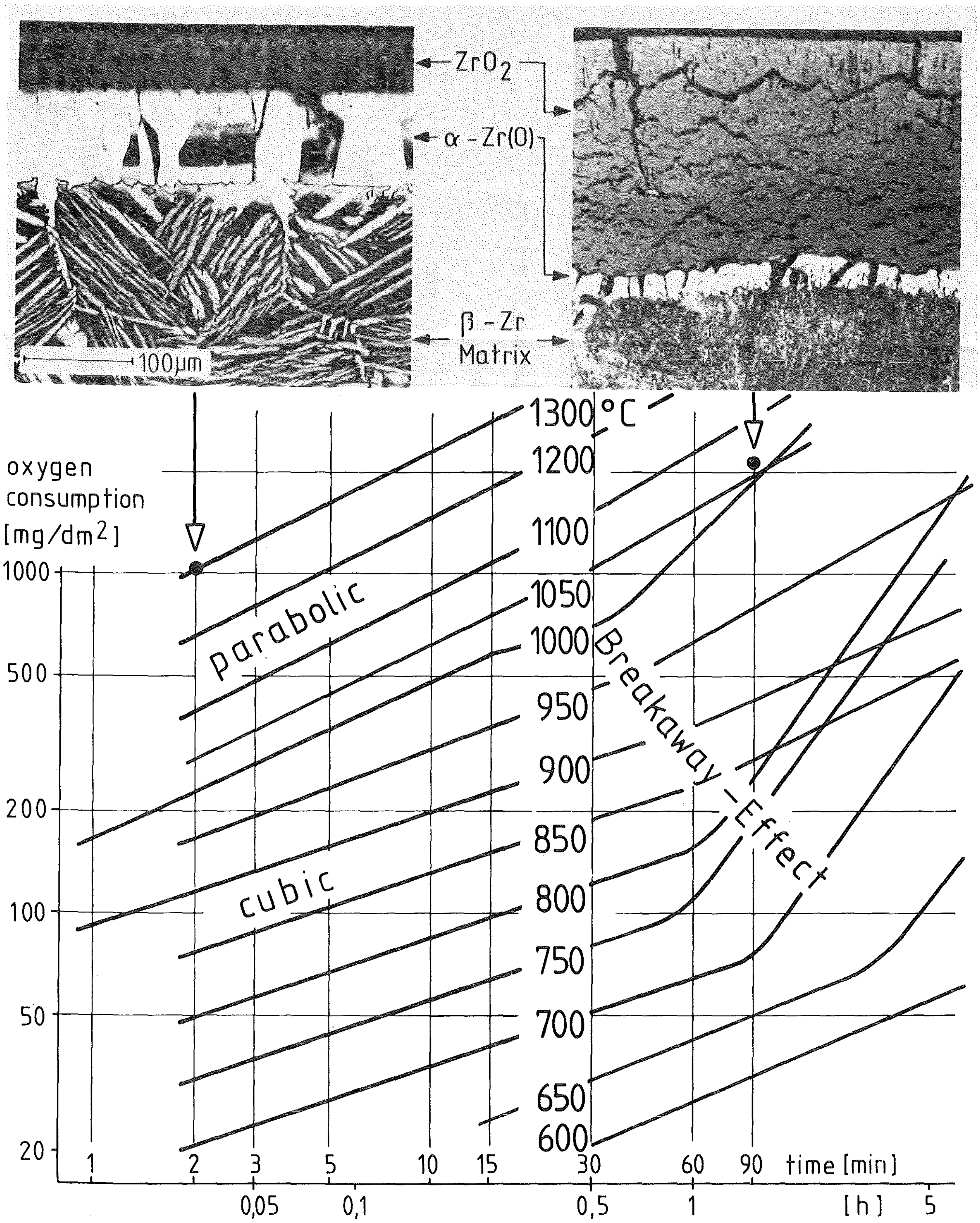


Fig. 13 LONG TERM COOLABILITY OF A PARTLY BLOCKED CORE

Fig. 14

Comparison of Experimental and SIMTRAN-CODE Calculated Results of Zircaloy 4/Steam-Oxidation Kinetics at High Temperatures

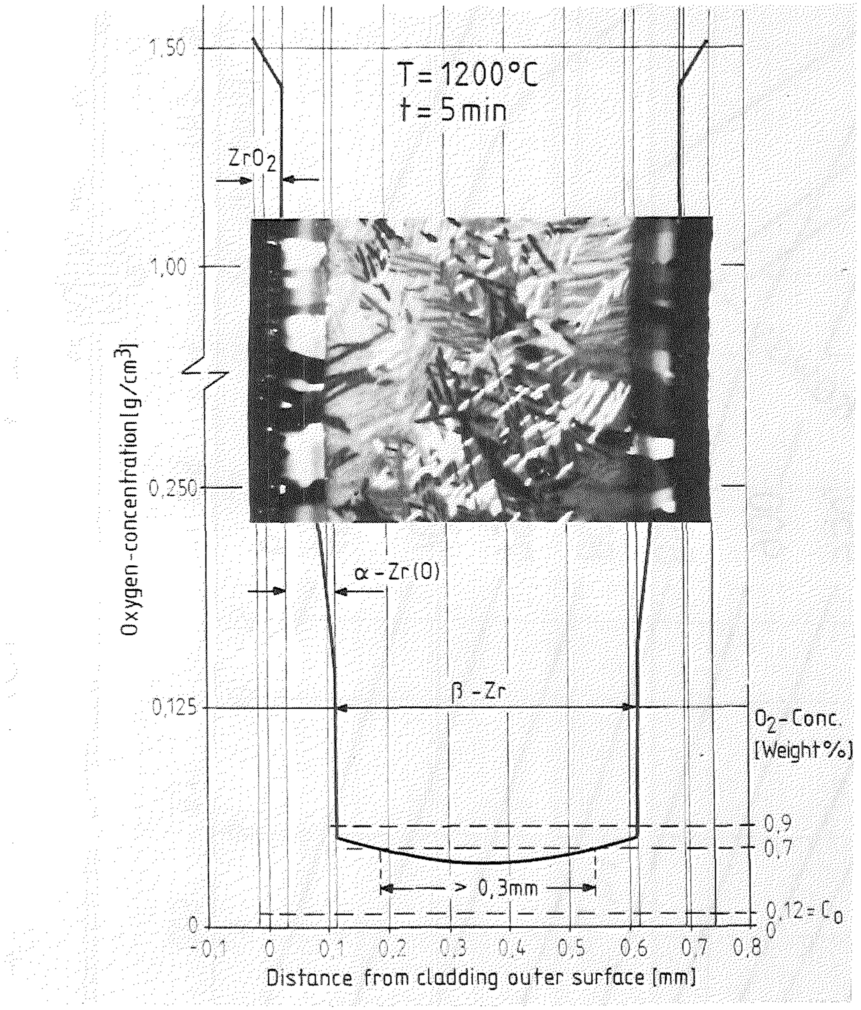




95 KfK IMF/II - PNS

Fig. 15

Kinetics of Zircaloy 4/Steam-High Temperature Oxidation with Respect to the "Breakaway-Effect"



A - At present valid embrittlement criterion:

Limitation of max. cladding temperature and of max. consumption of cladding wall by oxidation

$$T_{max} \leq 1200\ °C$$

$$s_{Zr\,(+O_2\,total} \xrightarrow{calc.} ZrO_2) \leq 17\%\, s_{wall}$$

B - Proposition of ANL:

Establishment of remaining ductile wall thickness

- with respect to stresses due to thermo-shocks during quenching:

$$s_{0,9\,Weight\%\,O_2} > 0,1\ mm$$

- with respect to handling, transport, intermediate storage of fuel elements:

$$s_{0,7\,Weight\%\,O_2} > 0,3\ mm$$

101 KfK IMF/II-PNS

Fig. 16

Penetration and embrittlement of Zircaloy cladding by steam oxidation: criteria of embrittlement

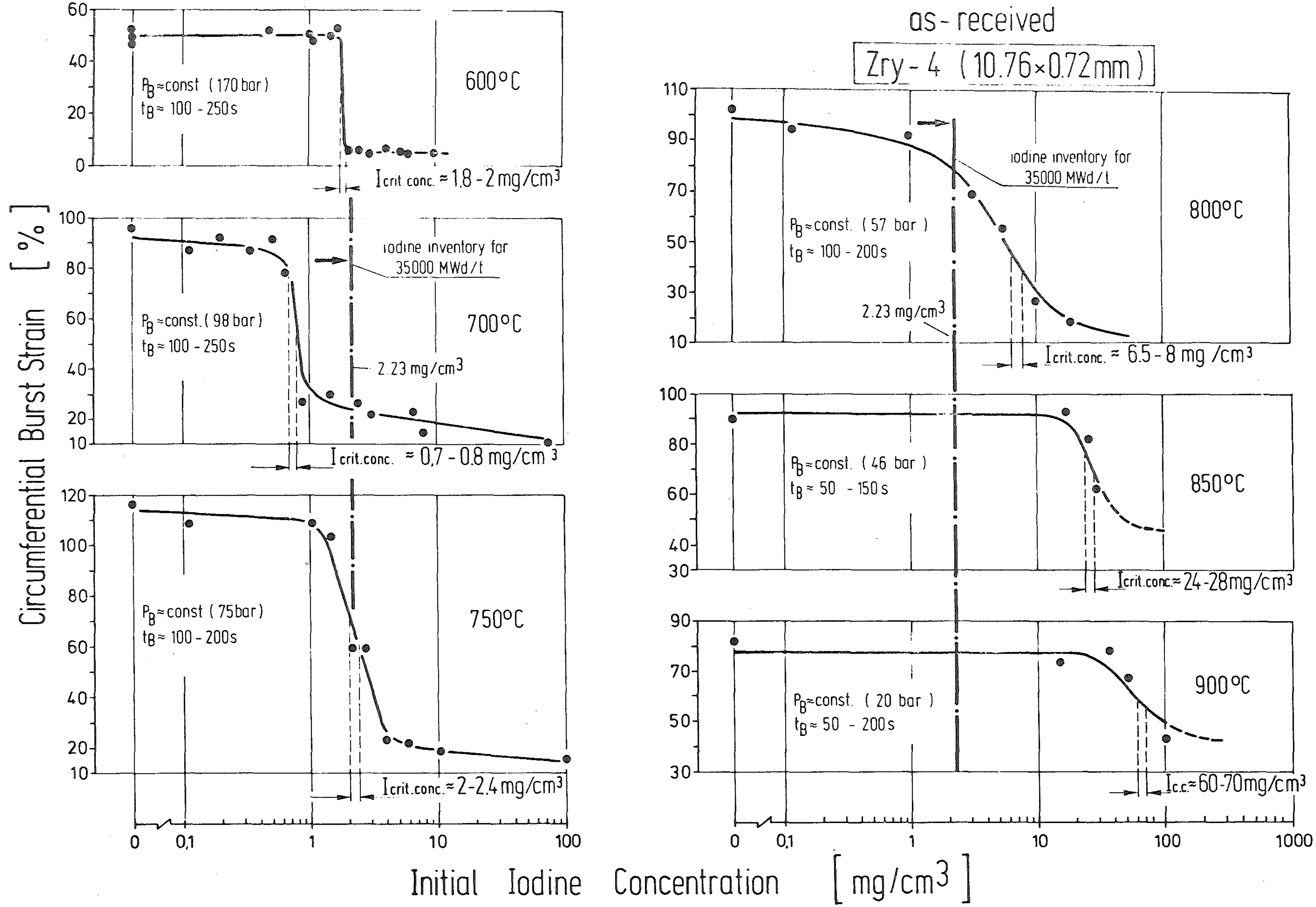


Fig. 17

Influence of the initial iodine concentration on the burst strain of as-received Zircaloy - 4 tubing between 600°C and 900°C

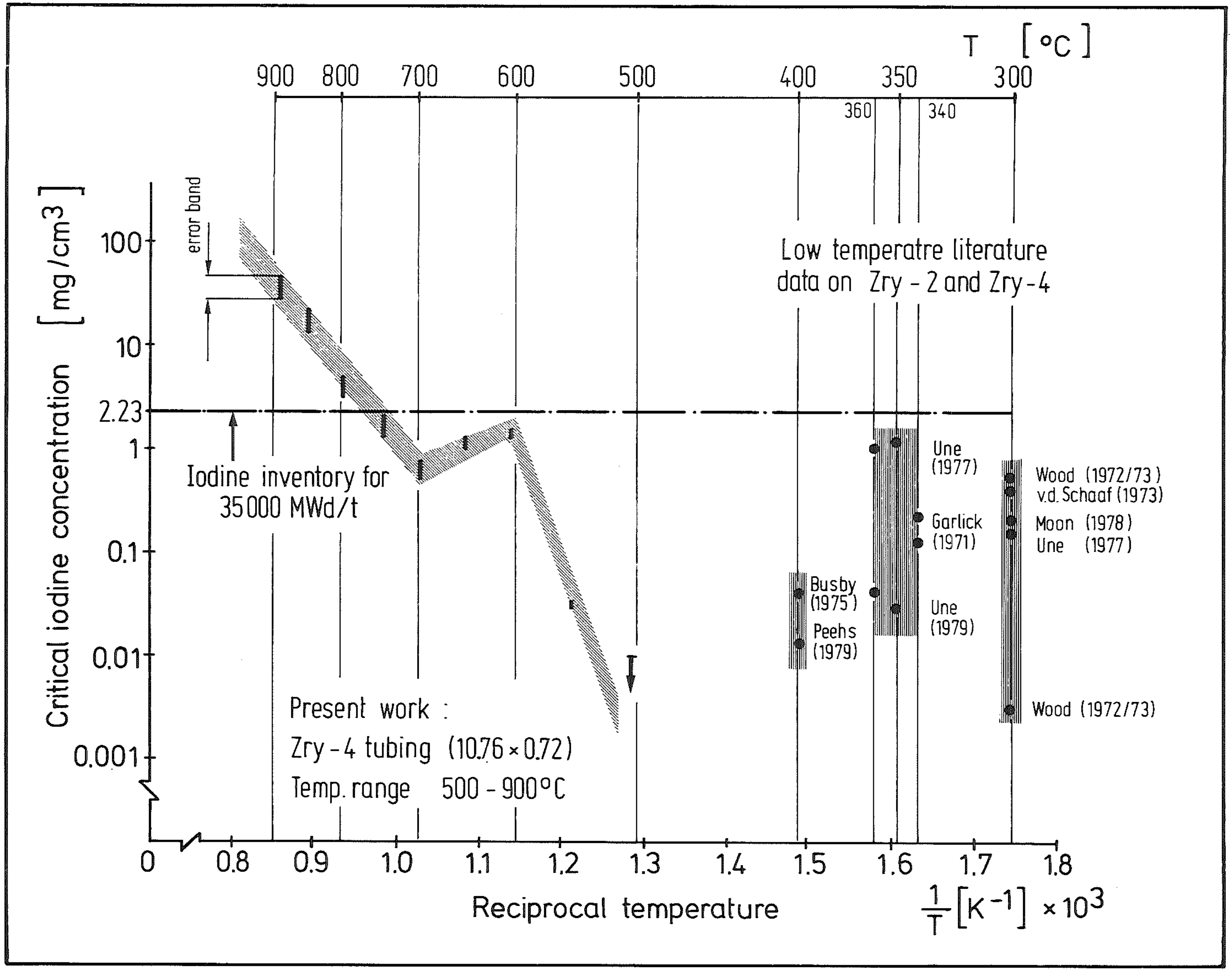


Fig. 18

IMF/I

Arrhenius plot of the critical iodine concentration resulting in SCC failure of as-received Zry - 4 tubing between 500 and 900°C

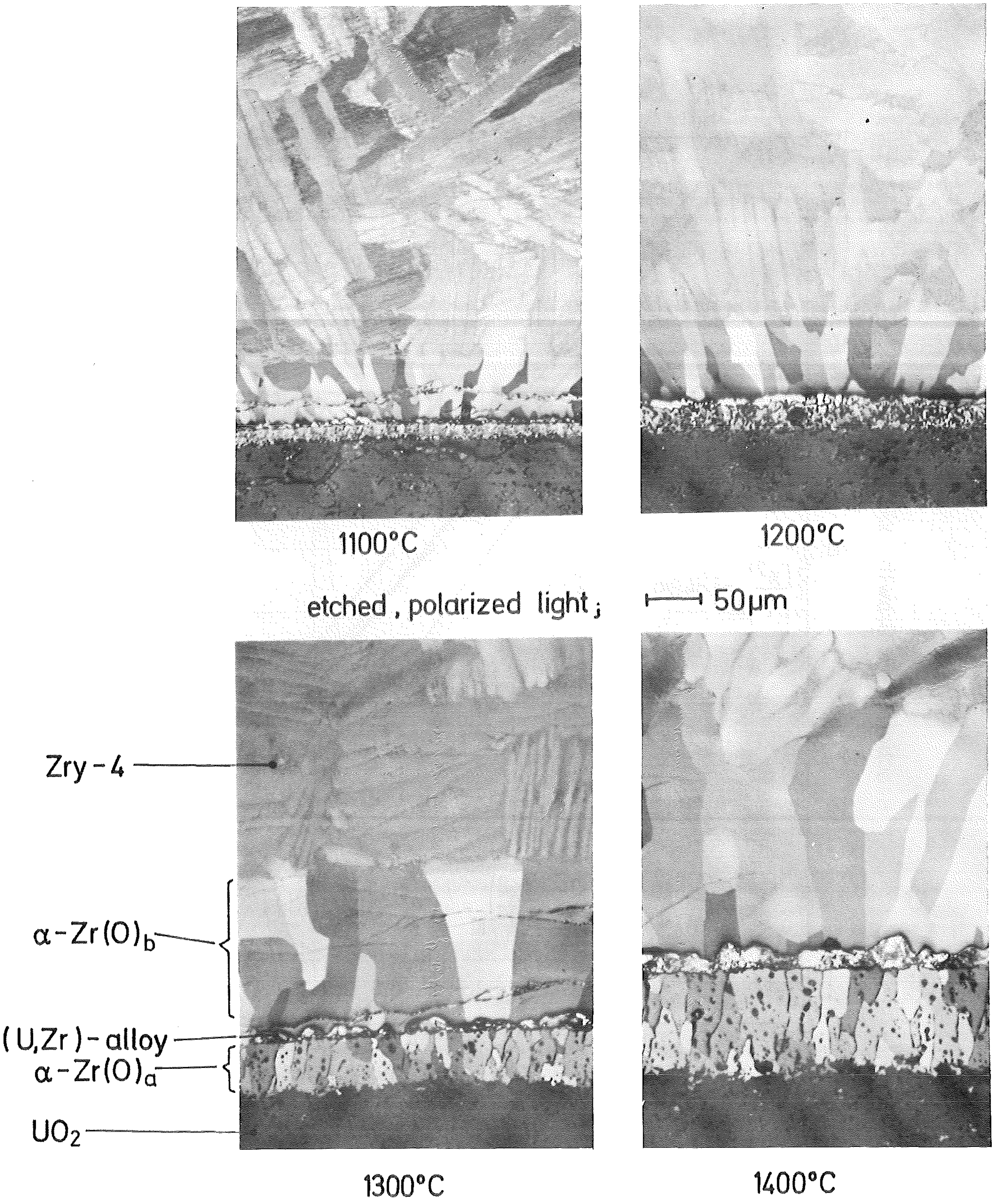


Fig. 19 UO_2/Zircaloy-4 interactions

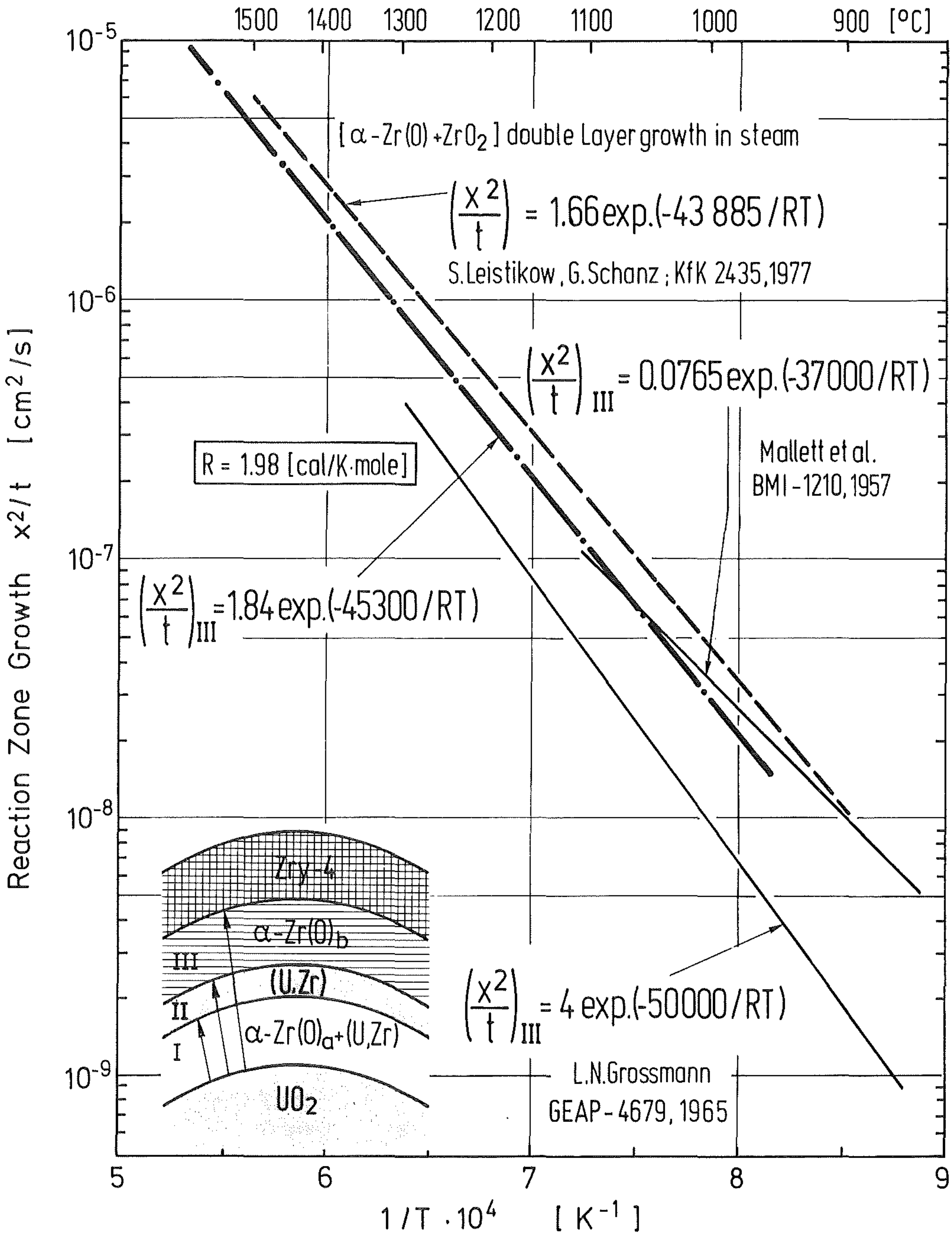


Fig. 20

Arrhenius plot of the combined UO_2/Zry reaction zones as determined by several investigations. Comparison with [α-Zr(O)+ZrO_2] double layer growth in steam

Element	Gap inventory[a] (% of total inventory)		Gap escape fraction (% of gap inventory)		Total release (% of total inventory)	
	WASH-1400	Model	WASH-1400	Model	WASH-1400	Model
Xe and Kr	8	1.27	100	100[b]	8	1.27[b]
Cs	15	2.79	33	0.89	5	0.025
I	10	2.79	33	1.91	3.3	0.053

[a] Calculated for stable and long-half-life isotopes. The gap inventory and total release of ^{133}Xe and ^{131}I would be 2 to 3 times lower.

[b] An additional amount of fission gas, approximately 1.5% of the total inventory, would be released during heatup.

Fig. 21

Comparison of WASH-1400 and model calculations (ORNL-results)

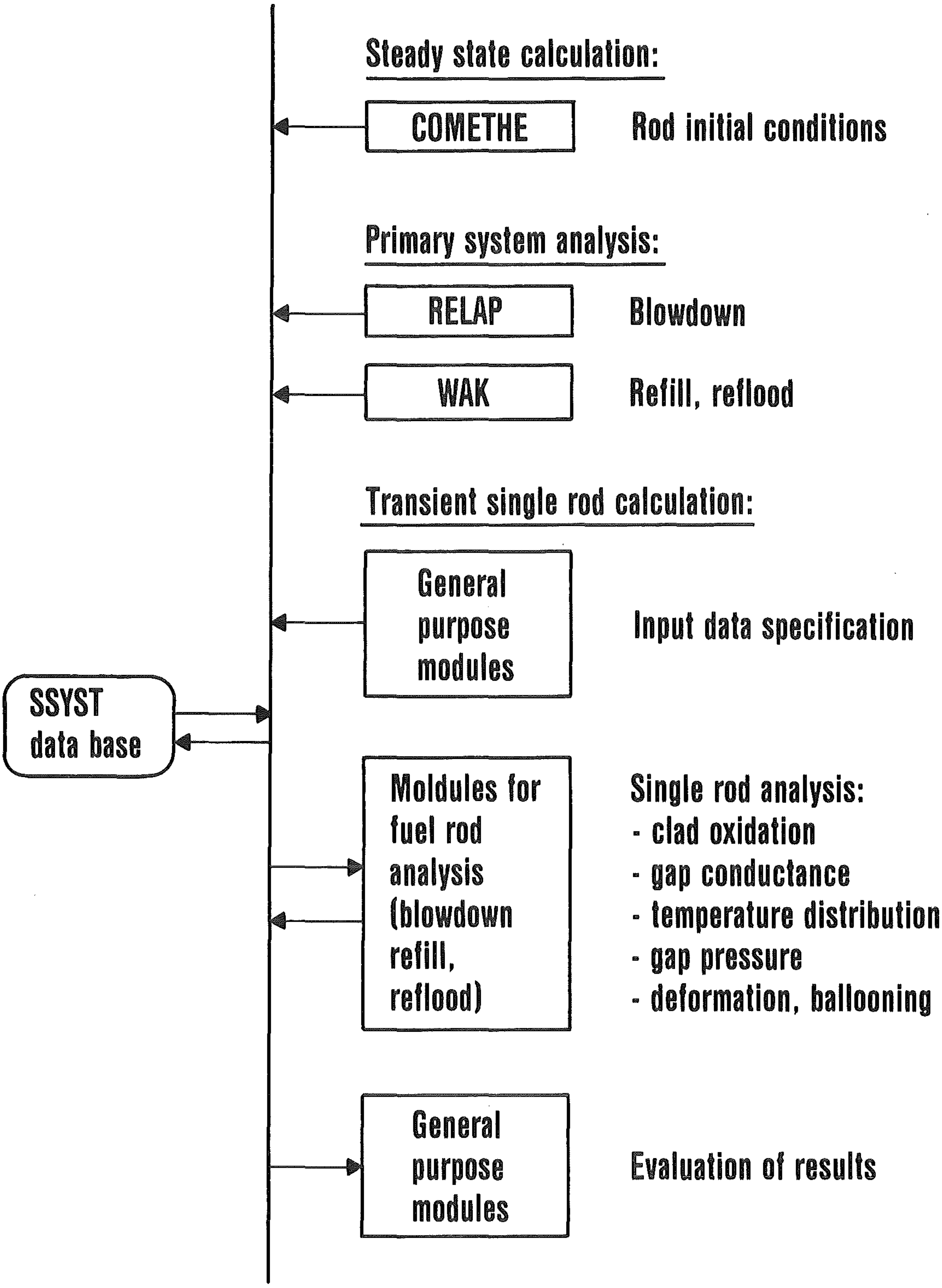


Fig. 22

SSYST-2: LWR Single Rod Analysis

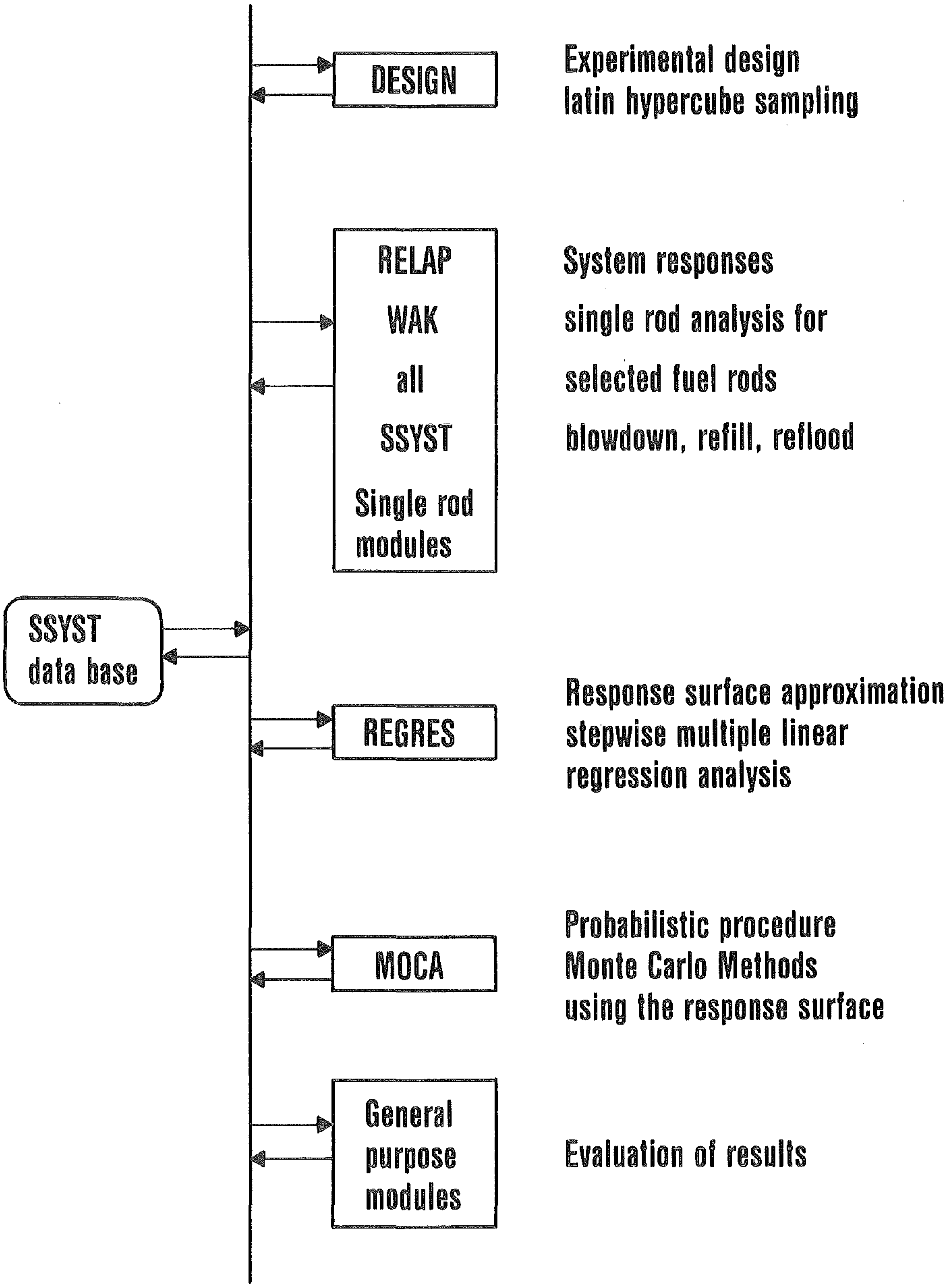
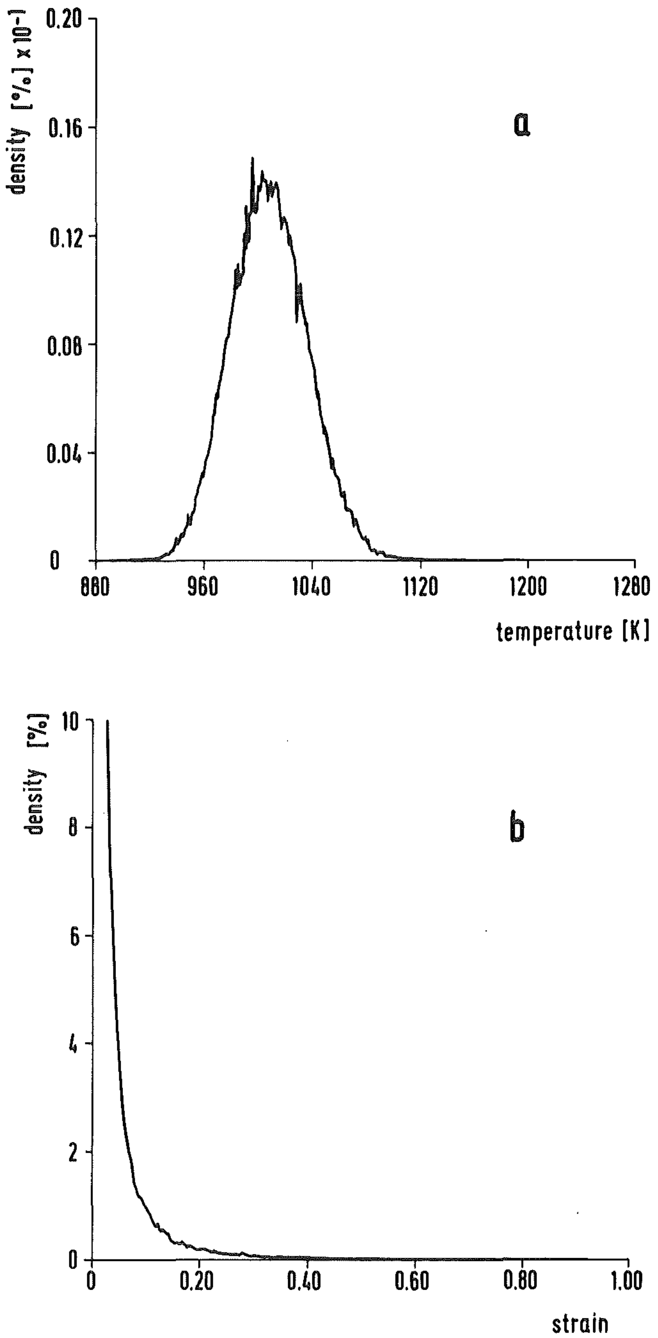


Fig. 23

SSYST-3: Probabilistic Methods to Estimate whole Core Damage



KfK IRE

Fig. 24

Probability density function of peak cladding temperature (a) and final strain (b)