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Heat Transfer in Rod Bundles with Severe Clad Deformations

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INSTITUT FÜR REAKTORBAUELEMENTE

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HEAT TRANSFER IN ROD BUNDLES WITH SEVERE CLAD DEFORMATIONS

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Abstract

The content of the paper is focused on heat transfer conditions during the reflood phase of a LOCA in slightly to severely deformed PWR fuel rod bundle geometries.

The status of analytical and, especially, of experimental work is described as far as it is possible within this frame. Emphasis is placed on the presentation of the results of "Flooding Experiments with Blocked Arrays" (FEBA), a program performed at the Kernforschungszentrum Karlsruhe in the frame of the Project Nuclear Safety (PNS).

Experiments performed out-of-pile show that coolant channel constrictions of up to 90 % do not lead to significant core coolability problems during reflood. This is even true for low water injection rates corresponding to a flooding velocity of 2 cm/s for the cold bundle.

The results of the thermal-hydraulic experiments cover rather widely the cladding temperature range below 1000 °C. However, outlining the total range of heat transfer conditions in severely damaged rod bundle geometries, investigations are mentioned performed in the frame of the PNS as well. They provide information about the condition of rod bundles being exposed to temperatures of up to 2000 °C prior to reflood.

Wärmeübergang in Stabbündeln mit schweren Hüllrohrverformungen

Zusammenfassung

Der Inhalt dieses Beitrages bezieht sich vornehmlich auf Wärmeübergangsbedingungen während der Flutphase eines Kühlmittelverluststörfalles in leicht bis schwer gestörten DWR-Brennstab-Bündelgeometrien.

Es wird der Stand der analytischen und insbesondere der experimentellen Arbeiten beschrieben, soweit dies in diesem Rahmen möglich ist. Der Schwerpunkt liegt auf der Darstellung der Ergebnisse aus "Flutexperimenten mit blockierten Anordnungen" (FEBA), einem Programm, das im Rahmen des Projektes Nukleare Sicherheit (PNS) im Kernforschunszentrum Karlsruhe durchgeführt worden ist.

Außerhalb des Reaktors durchgeführte Experimente zeigen, daß beim Fluten eines Reaktorkerns lokale Kühlkanalverengungen bis 90 % nicht zu nennenswerten Kühlproblemen führen. Dies gilt selbst für kleine Wassereinspeiseraten, die einer Flutgeschwindigkeit von 2 cm/s im kalten Bündel entsprechen.

Der Temperaturbereich unter 1000 °C für die maximalen Hüllrohrtemperaturen wird durch die Ergebnisse der thermohydraulischen Experimente umfassend abgedeckt. Um jedoch den Gesamtbereich der Wärmeübergangsbedingungen in stark beschädigten Stabbündelgeometrien zu umreißen, werden auch Untersuchungen angeführt, die ebenfalls im Rahmen des PNS durchgeführt werden. Sie liefern Informationen über den Zustand von Stabbündeln, die Temperaturen von bis zu 2000 °C vor dem Fluten ausgesetzt waren.

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1. INTRODUCTION

The discussion of "Thermalhydraulics of Degraded Core Condition" presented in this paper is concentrated on pressurized water reactor (PWR) problems during a loss-of-coolant accident (LOCA). Moreover, emphasis is placed on the heat transfer conditions in slightly to severely deformed fuel rod bundle geometries. The emergency core cooling systems (ECCS) are usually assumed to provide sufficient capacity to avoid significant fuel damage for current design basis accident cases. However, during such an event situations may arise leading to the heatup of the core for an extended time span. Under design basis accident conditions this may be the case, e.g. during the refill phase after blowdown. It is assumed that the core is heated up by the nuclear decay heat with ineffective cooling by superheated steam. The reflood water injection has to terminate this "Second Heatup Phase" before fuel rod damage exceeds specified limits.

The study of the conditions leading to initiation of core degradation, therefore, requires knowledge of the processes involved in core cooling beyond the limits specified for a LOCA. The coolability of a partly degraded core determines the continuation or termination of the sequence of further degradation as well as the extent of fission products releasing.

There are many paths for a sequential degradation of the core beyond design basis loss-of-coolant accidents depending on the initial conditions, the individual system behavior of the plant, and the different interventions by emergency cooling. Since a complete analysis of such sequences is difficult, investigations are under way assuming complete core melt down and analyzing the consequences for the case of no successful intervention. However, in the risk analysis and the evaluation of the effectiveness of emergency cooling of degraded cores, the coolability of the different core configurations should be more precisely known in order to terminate the disruptive event as effectively as possible.

To assess the problems of heat transfer mechanisms possibly leading to fuel rod cladding deformation and those of degraded core cooling, analytical as well as experimental investigations are under way. Contributions known to the author are listed in /1 through 41/.

The most serious LWR core cooling problems, e.g. during a LOCA reflood, are found in the dispersed flow portion which is characterized by very small water content in a significantly superheated steam. The complex heat transfer processes taking place between the two phases and hot walls in deformed rod bundle geometries determine whether or not the core will be coolable. An experimental investigation of the influence of severely deformed fuel rod claddings on reflood heat transfer was subject e.g. of the FEBA program /11, 22, 39/. The results presented include blockage ratios of up to 90 % of the coolant channel cross sections, cladding temperatures up to 1000 ^{o}C and flooding velocities as low as 2 cm/s.

Within the scope of this paper, it is not suitable to give a complete description of all possible sequences of events after initiation of fuel cladding deformation. Therefore, only three main incidents are considered which could lead to fuel rod damage as well as core degradation.

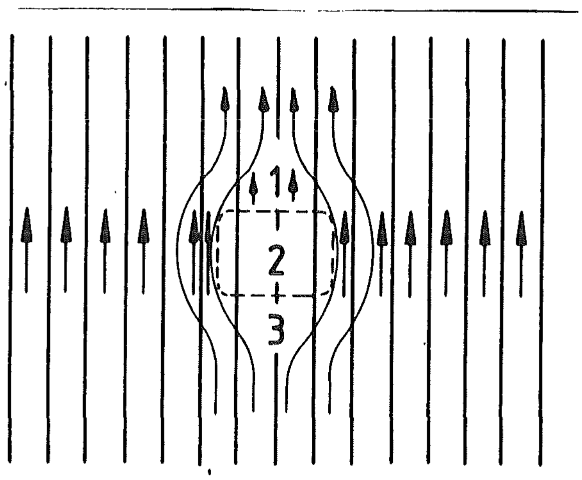
In Section 2 the conditions during the early phase of core deformation during a LOCA are described. The conditions at termination of core deformation - due to any transient - characterized by long-term cooling of severely deformed fuel rod claddings are described in Section 3, and those of fuel debris in Section 4. For the conditions at beginning of severe fuel bundle damage and core disruption remain open questions.

2. REFLOOD BEHAVIOR OF A CORE WITH EXTENDED CLAD BALLOONING AT LOW FLOODING VELOCITIES

Early termination of a LOCA requires reflooding the core on time by a sufficiently fast rise of the water level in the core. After depressurization of the primary cooling system and during partial or complete dryout of the reactor vessel, some fuel rod claddings may balloon due to internal overpressure and increasing cladding temperatures. The most serious deformation then is expected in the center of the core. The number of deformed adjacent rods as well as the amount of ballooning increase for decreasing ECC water injection rates. Moreover, such local balloons reduce the subchannel cross sections and influence the fluid flow and heat transfer during the reflood phase following. On the other hand, if the deformation of the cladding continues during reflood, the final size and shape of these coolant channel blockages are in turn to be influenced by the local cooling conditions.

The effect of coolant channel blockages by ballooned fuel rods on reflood heat transfer has been investigated in several out-of-pile reflood experiments, e.g. FLECHT/SEASET /11, 34/, THETIS /35/, 2D /12/ and FEBA /39/. The results from these experiments are being used for computer code modeling, e.g. COBRA TF /13/ and SCDAP /14, 15/, as well as for complementing analyses of out-of-pile and in-pile fuel rod behavior tests under LOCA conditions /16, 17, 18, 19, 20, 21, 31, 32/.

The overall hydraulic conditions of a PWR core can be roughly characterized by the upward flow of the coolant in a number of parallel vertical subchannels. If in the center of such an array some of the coolant subchannels are constricted, the fluid mass flow through the blocked portion is then reduced compared with the average mass flow of all the subchannels. The portion of the core being immediately affected by the decreased coolant flow can be subdivided into three main regions as indicated in Fig. 1:



1 Droplet fall back

2 Reduced mass flux to which extent?

3 Droplet deentrainment?

Fig. 1 SKETCH OF A PWR CORE WITH REFLOOD FLOW DIVERSION AT COOLANT SUBCHANNEL CONSTRICTIONS IN THE CENTER

Examples of some of the main effects which influence reflood cooling in rod bundles with coolant channel constrictions about the bundle midplane, investigated in the FEBA-program /39/, are presented in the following:

2.1 Effect of Blockage Ratio

Figure 2 shows a clad temperature transient at an elevation 10 mm downstream of a flow blockage placed at the bundle midplane and a corresponding transient in the bypass area. Ballooned fuel rod claddings are simulated by sleeves constricting the individual subchannel cross sections of a 3x3 rod cluster by 62 % as indicated on the sketch on the figure.

The test was performed with an electrically simulated decay heat power of 120 % ANS-Standard, a system pressure of p = 4 bar, and a forced feedwater injection corresponding to a velocity of v = 3.8 cm/s for the rising water level in the cold bundle.

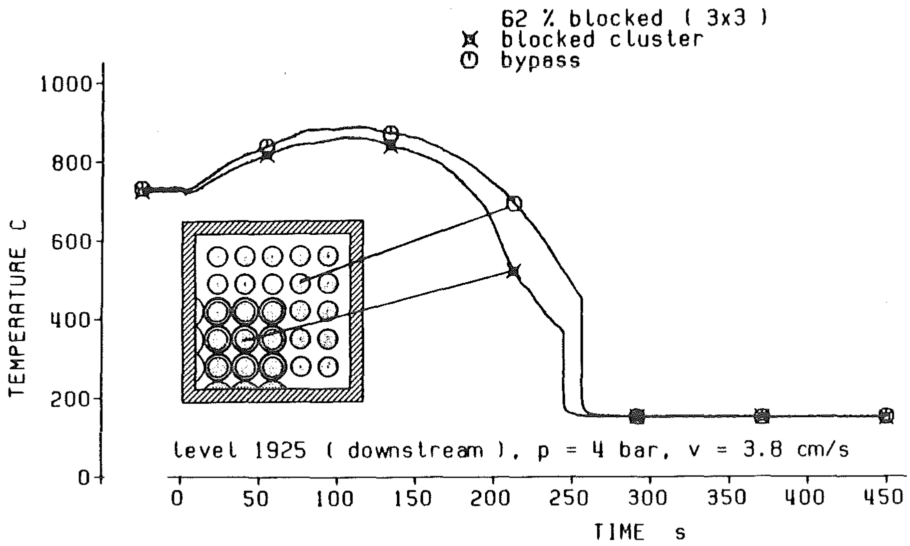


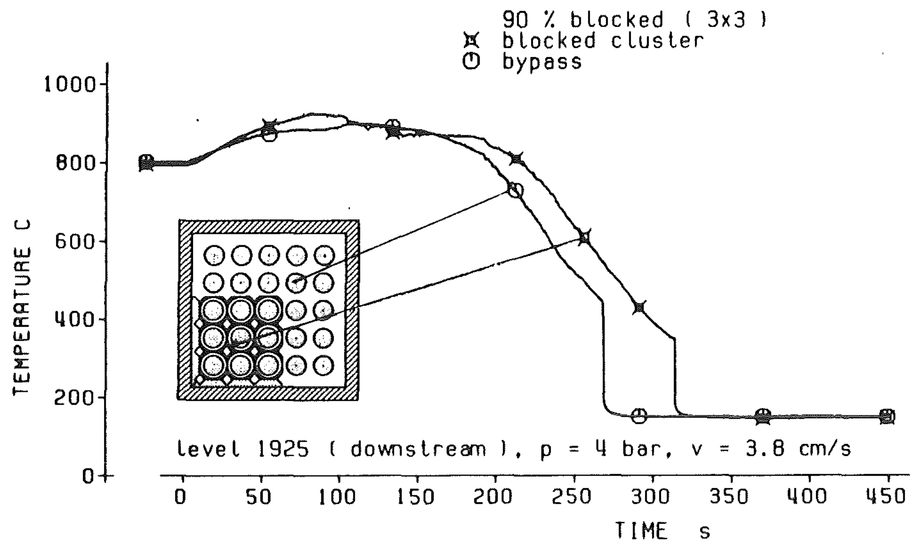
Fig. 2 CLADDING TEMPERATURES 10 mm DOWNSTREAM OF A 62 % BLOCKAGE AND IN THE BYPASS

A blockage causes two opposite effects on the local reflood heat transfer:

- Within and downstream of the blocked portion of the bundle, the coolant mass flux is reduced which may lead to reduced local cooling.
- Two-phase flow passing through a blockage may lead to improved cooling due to enhancements of turbulence and water droplet dispersion.

For the geometry and the reflood conditions indicated, the two-phase cooling enhancement effect overshadows the coolant mass flux reduction. This is demonstrated in the comparison of the transients plotted in Fig. 2. Downstream of the blockage, the cladding temperatures are lower than at the same axial level in the bypass area.

Increasing the blockage ratio reverses the relative influence of the two afore mentioned effects for the same reflood conditions as shwon in Fig. 3. The clad temperature transients were obtained



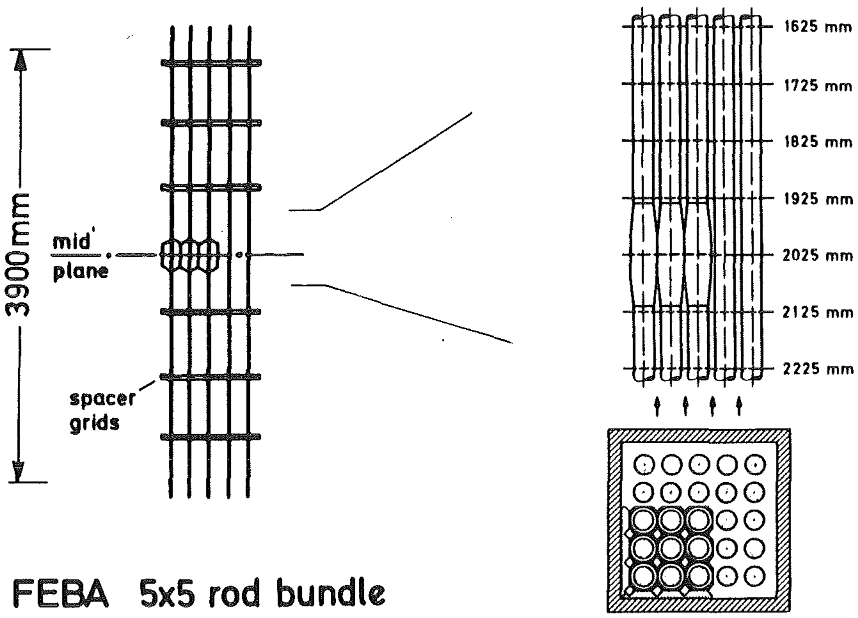


Fig. 3 CLADDING TEMPERATURES 10 mm DOWNSTREAM OF A 90 % BLOCKAGE AND IN THE BYPASS

from the same measurement locations during a test performed with identical reflood conditions as mentioned for Fig. 2. However, the blockage was constricting the subchannel cross section by 90 %. Under these conditions, the coolant mass flux reduction overshadows the two-phase cooling enhancement effect at blockages. Downstream of the blockage, the cladding temperatures are slightly higher than at the corresponding elevation in the bypass area during most of reflood. With the high blockage ratio taken into account, the temperature rise downstream of the blockage is moderate.

The temperature of the cladding downstream of the blockage decreases at a slower rate than that in the bypass and quenching of the rods occurs significantly later. Thus, the heat removal there is delayed. However, during this period of the reflood transient, the influence of the temperatures on cladding deformation is relatively unimportant. Most of deformation takes place during refill and the very beginning of reflood /16, 17/.

For an assumed blockage ratio of up to about 60 % due to cladding deformation during the refill phase, the reflood cooling downstream of the blocked portion of the rod cluster appears to reduce the probability for further deformation in this region in comparison with the situation in rod clusters without deformation. The cladding temperatures there become generally low due to the effects of enhanced turbulence and droplet dispersion as shown in Fig. 2.

For a blockage ratio of about 90 %, the downstream cladding temperatures are not significantly higher than those for unblocked rod clusters at corresponding axial elevations as shown in Fig. 3. Therefore, at high blockage ratios the presence of blockages would not unduely promote further axial propagation of cladding deformation in the downstream region.

For an analysis of the heat transfer mechanisms within the blocked portion of a rod cluster with highly ballooned claddings, more detailed descriptions would have to be made about the geometry of the balloons. The blockage configurations shown in Figs. 2 and 3 represent coplanar balloons of identical shape surrounding the individual rods in a concentric fashion. No burst openings of the claddings e.g. are simulated. This array allows better instrumentation and easier data analysis for thermal-hydraulic experiments as compared to those with complex geometries resulting from actual cladding deformation tests. With the thermal behavior of the sleeves as previously mentioned, simulation of ballooned fuel rod cladding is a compromise with the following properties: the stagnant steam filled gap of 0.8 mm width between outer surface of heater rod and inner surface of the sleeve assumes a gap coefficient which is of the same order of magnitude as that of ballooned fuel rods. For the flooding conditions applied in the tests (v = 2 through 10 cm/s, p = 2 through 6 bar), the 1 mm thick sleeve walls do not produce significantly different temperature transients from those with ballooned Zr4-claddings of fuel rod simulators. With respect to the quench time within the blokkage, the 1 mm thick wall leads to rather conservative results /22, 23/.

Some additional characteristics of the sleeve blockage array will also have to be mentioned. The sleeve and wall blockage devices constrict the subchannels for a length of 125 mm for the 62 % blockage and of 65 mm for the 90 % blockage. For a given blockage ratio, the length of the severely constricted subchannels causes the throttling effect of a blockage as well as the volume of the bundle portion, which has to be cooled by reduced coolant mass flux.

Surface temperature transients measured at the axial midplane of a 62 % sleeve blockage array and at the corresponding elevation in the bypass of the blockage are shown in Fig. 4. The data are obtained from the same test as those plotted in Fig. 2. The temperature of the rod portion underneath the sleeve rises faster at the beginning of reflood than that of a corresponding rod portion in the bypass, indicating reduced heat removal from the rod portions within the blockage. However, the temperature of the sleeve decreases rapidly during this period.

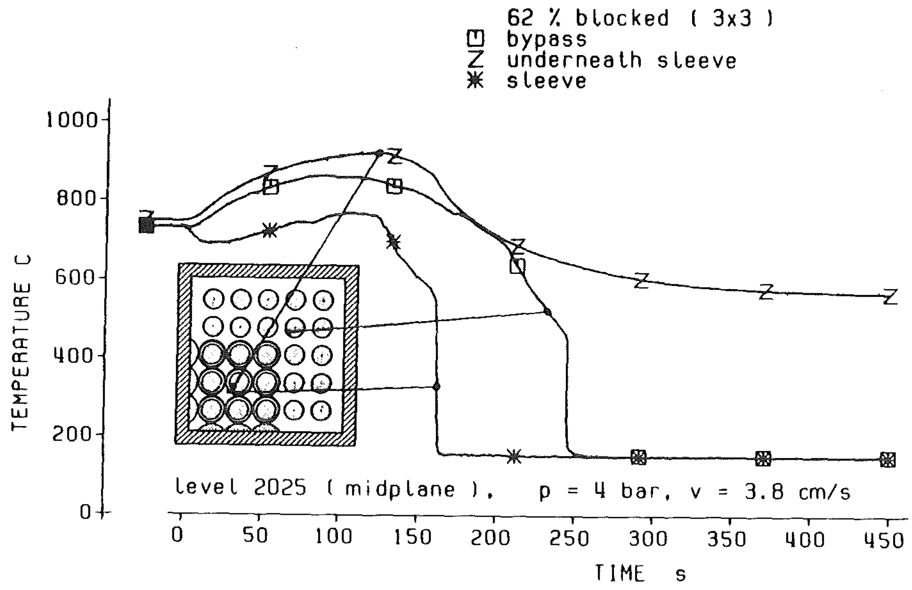


Fig. 4 TEMPERATURES AT THE SURFACES OF A SLEEVE, OF A ROD COVERED BY A SLEEVE, AND OF A ROD IN THE BYPASS OF A 62 % BLOCKAGE

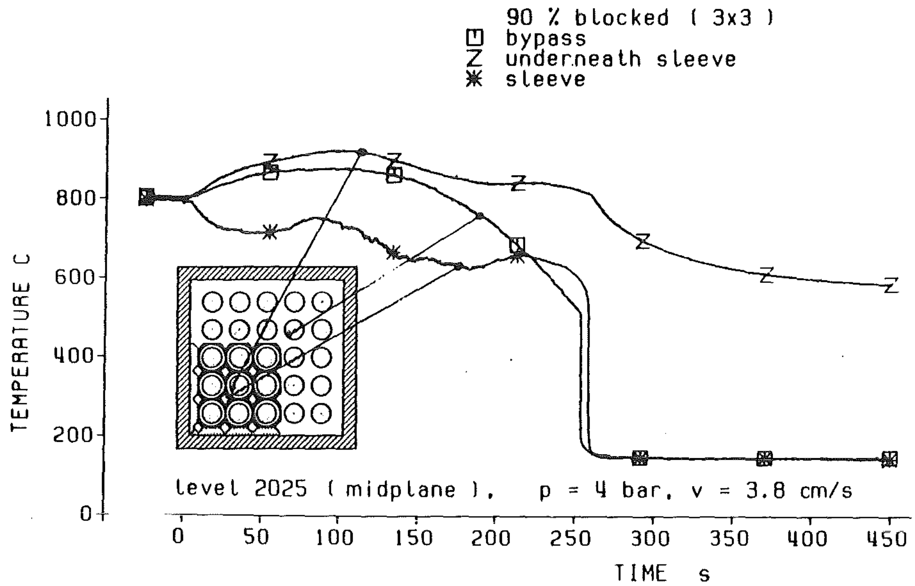


Fig. 5 TEMPERATURES AT THE SURFACES OF A SLEEVE, OF A ROD COVERED BY A SLEEVE, AND OF A ROD IN THE BYPASS OF A 90 % BLOCKAGE

The reasons for this behavior are the following:
The rods with unballooned as well as with ballooned claddings are heated up during the refill phase of ineffective cooling to approximately the same temperatures. The cooling is enhanced rapidly when the rising water level reaches the heated portion of the core at the lower end of the rods. The resulting steam with entrained water droplets flows upwards through the bundle with a velocity in the range of 10 to 30 m/s at the upper end of the rods. Being partially insulated by the sleeves at the blockages, the rods underneath are heated up faster than those being in better contact with the cladding which is exposed to the enhanced cooling after the onset of the reflood flow. Moreover, the coolant mass flux through the constricted subchannel is reduced. But, it is sufficient to cool the sleeves down to significantly lower temperatures than the claddings in the bypass area. Once a sufficient temperature difference is established between the sleeve and the rod surface underneath, an increasing amount of the heat produced is transmitted through the gap. After the turnaround point of the rod temperature, even portions of the stored heat are removed in addition. With continued increase of reflood cooling, the sleeve is being quenched leading to increased heat transmission across the gap. The quench front in the bundle at this time is still about 700 mm upstream and it arrives approximately 80 s later at the blockage elevation quenching then all rod portions around the blockage except those covered by the sleeves. At the lower end of the sleeves the quench front on the rods stops inspite of the fact that the sleeves are quenched already. Downstream of the sleeves, a new quench front has to be initiated at the rod surfaces. The axial heat conduction in the rods is not sufficient to remove the heat produced underneath the sleeves. Most of this heat is transmitted radially across the gap for high as well as low flooding velocities (see Section 2.2). A heat transfer analysis is given in Section 2.4.

The situation in the case of 90 % blockage is not very different from that in the case of 62 % blockage previously discussed. Fig. 5 shows the plots of three temperature transients in a test of 90 % blockage measured at the very same axial elevations as in the test of 62 % blockage with identical flooding conditions as shown in Fig. 4. During the initial period of flooding, the sleeve and the rod surface underneath as well as the cladding of the rod in the bypass register temperature transients very similar to those of the 62 % blockage. However, the sleeves of the 90 % blockage do not quench before the quench front reaches the blockage elevation. Since for the 90 % blockage a much smaller part of the surface of the sleeve is exposed to the coolant than for the 62 % blockage, the coolant mass flux through the severely constricted subchannel is not able to quench the sleeve ahead of the arrival of the main quench front. It is sufficient, however, to remove the heat produced in the portions of the rods covered by the sleeves, leading to turnaround and decreasing temperatures.

For the tests reported in Figs. 2 through 5, the cladding temperatures upstream of the blockage are the same for comparable elevations in the bypass region for the same flooding conditions (data not shown). Somewhat decreased cooling upstream of the blockages is observed for a much lower flooding velocity of 2.2 cm/s (see Section 2.2).

2.2 Effect of Flooding Velocity

In the 62 % blockage array with bypass, a test was performed with a flooding velocity of 2.2 cm/s. Temperature transients in the blocked rod cluster and in rods placed in the bypass area are compared for several axial elevations (see Fig. 6). Upstream of the blockage the cladding temperatures are slightly higher than in the bypass area. The reduced vapor and droplet mass fluxes due to flow diversion are believed to be mostly responsible for this effect. During the most critical mist flow portion of reflood, the sleeve temperature is lower than the cladding temperature at the same level in the bypass region inspite of the mass flux reduction. However, there is no earlier quenching of the sleeve as observed for the test performed with a flooding velocity of 3.8 cm/s. The general thermal behavior of the sleeve is more similar to that observed in the test performed with the 90 % blockage and a flooding velocity of 3.8 cm/s.

Downstream of the blockage, the same peak cladding temperatures occur as measured in the bypass region at the same elevation. The heat removal there is somewhat delayed. A new quench front on the rods has to be initiated downstream of the blockage after the quench front of the unblocked rod cluster has already

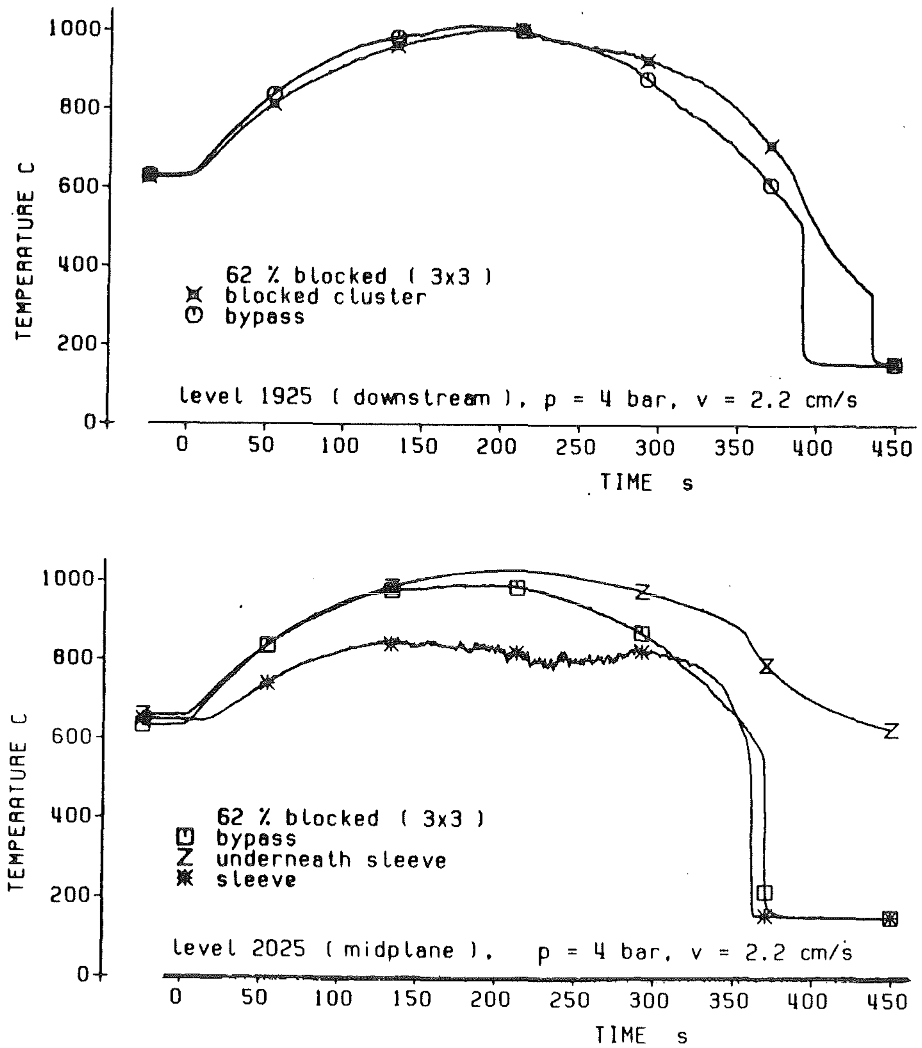


Fig. 6 COMPARISON OF CLADDING TEMPERATURES MEASURED IN THE BLOCKED AND IN THE BYPASS REGION OF A 62 % BLOCKAGE

passed. On the other hand, for the tests performed with flooding velocities of 3.8 cm/s and higher, the new quench front there is initiated ahead of the bypass quench front /11/. This behavior is true for sleeve blockage tests. It remains to be shown to what extent it is valid for ballooned fuel rod claddings.

For the partial blockages presented up to this point the true coolant mass flux through the constricted subchannels is unknown. Fig. 7 shows temperature transients measured in a test performed with identical conditions as those for the test reported in Fig. 6.

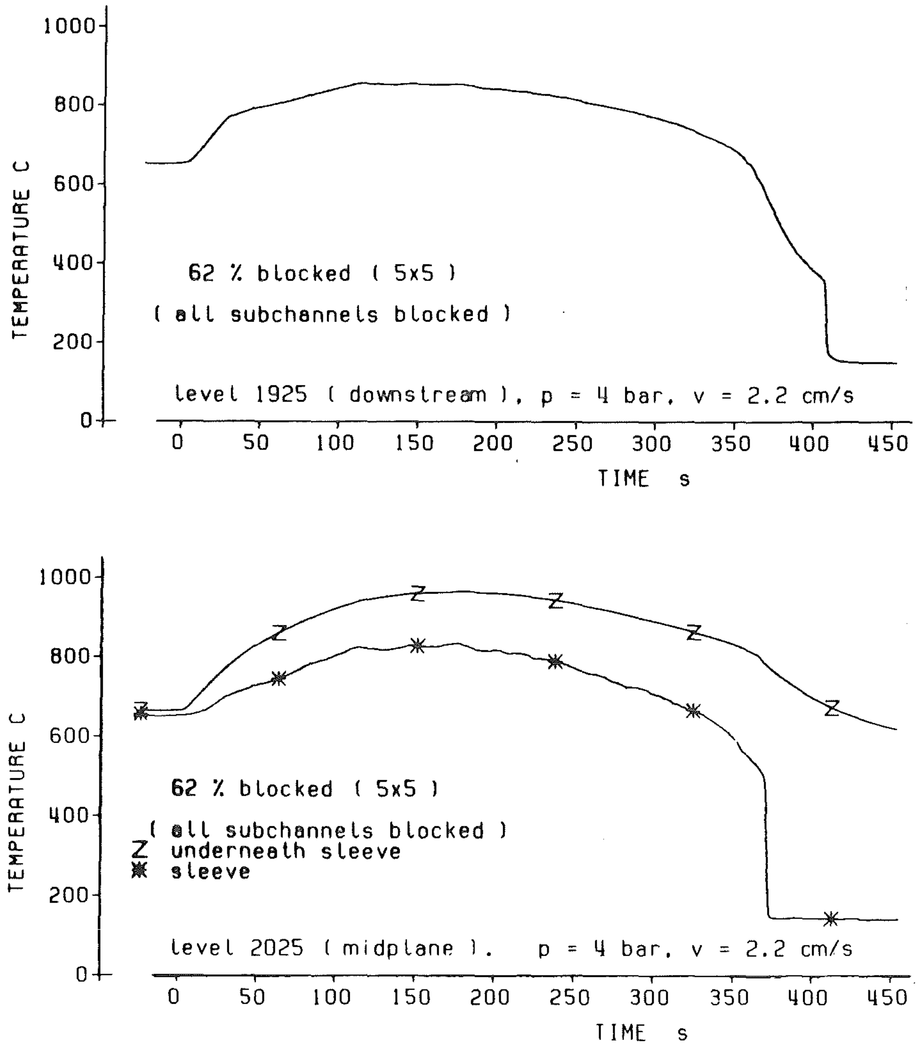


Fig. 7 SURFACE TEMPERATURES AT A 62 % BLOCKAGE FOR ALL 36 SUB-CHANNELS IN A 5x5 ROD BUNDLE

However, all subchannels are blocked in this case. The mass flux transient through the constricted subchannels corresponds then to the unblocked bundle mass flux. Important differences occur comparing the cladding temperature transients measured downstream of both the blockages as demonstrated in Fig. 8a. Significantly different maximum temperatures occur. For the case without bypass of the blockage, the turnaround temperature is about 150 K lower than that of the case with blockage bypass. The quench times are not as different from each other as one would expect from the highly different precursory cooling conditions. A surprising result is the fact that the conditions within this blockage configuration are very similar to those of the 62 % partial blockage. A comparison of the sleeve temperatures obtained from the two different tests is shown in Fig. 8b. For the partial blockage the sleeve temperature is slightly higher during the last period only.

The pressure drop along the blockage constricting all 36 subchannels of the bundle is lower than 0.01 bar during the mist flow regime and for the partial blockage it is again even slightly lower (data not shown). There are some similarities between the effect of such a blockage and that of a grid spacer /24, 25/.

Figure 9 shows typical examples of two-phase flow cooling enhancement downstream of different flow obstacles for identical transient coolant mass fluxes in all subchannels. The peak cladding temperatures are reduced significantly for increased size and shape of the flow obstacles. This comparison verifies the results obtained for blockages with bypass indicating more or less compensation of coolant mass flux reduction through blockages by enhanced cooling effectiveness of the remaining coolant mass flux.

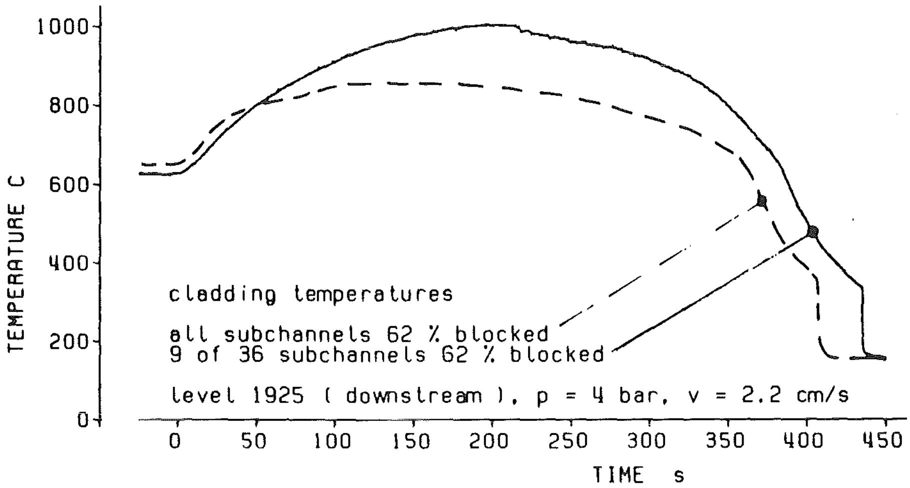


Fig. 8a COMPARISON OF CLADDING TEMPERATURES DOWNSTREAM OF THE PARTIAL BLOCKAGE WITH THOSE OF THE ARRAY BLOCKING ALL SUBCHANNELS

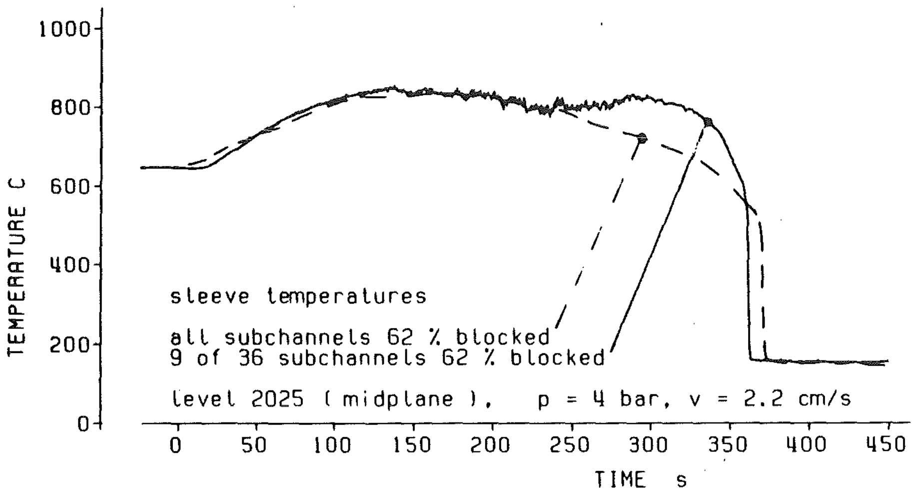


Fig. 8b COMPARISON OF SLEEVE TEMPERATURES OF THE PARTIAL BLOCKAGE WITH THOSE OF THE ARRAY BLOCKING ALL SUBCHANNELS

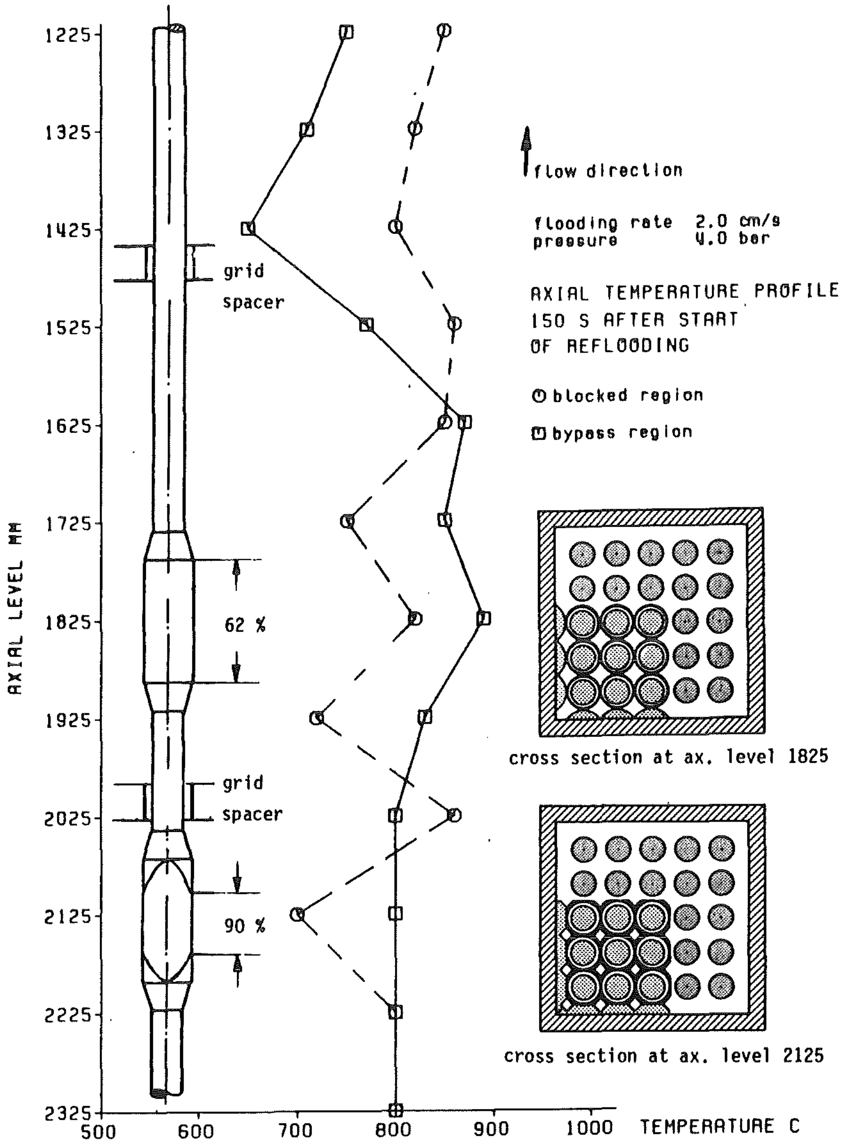


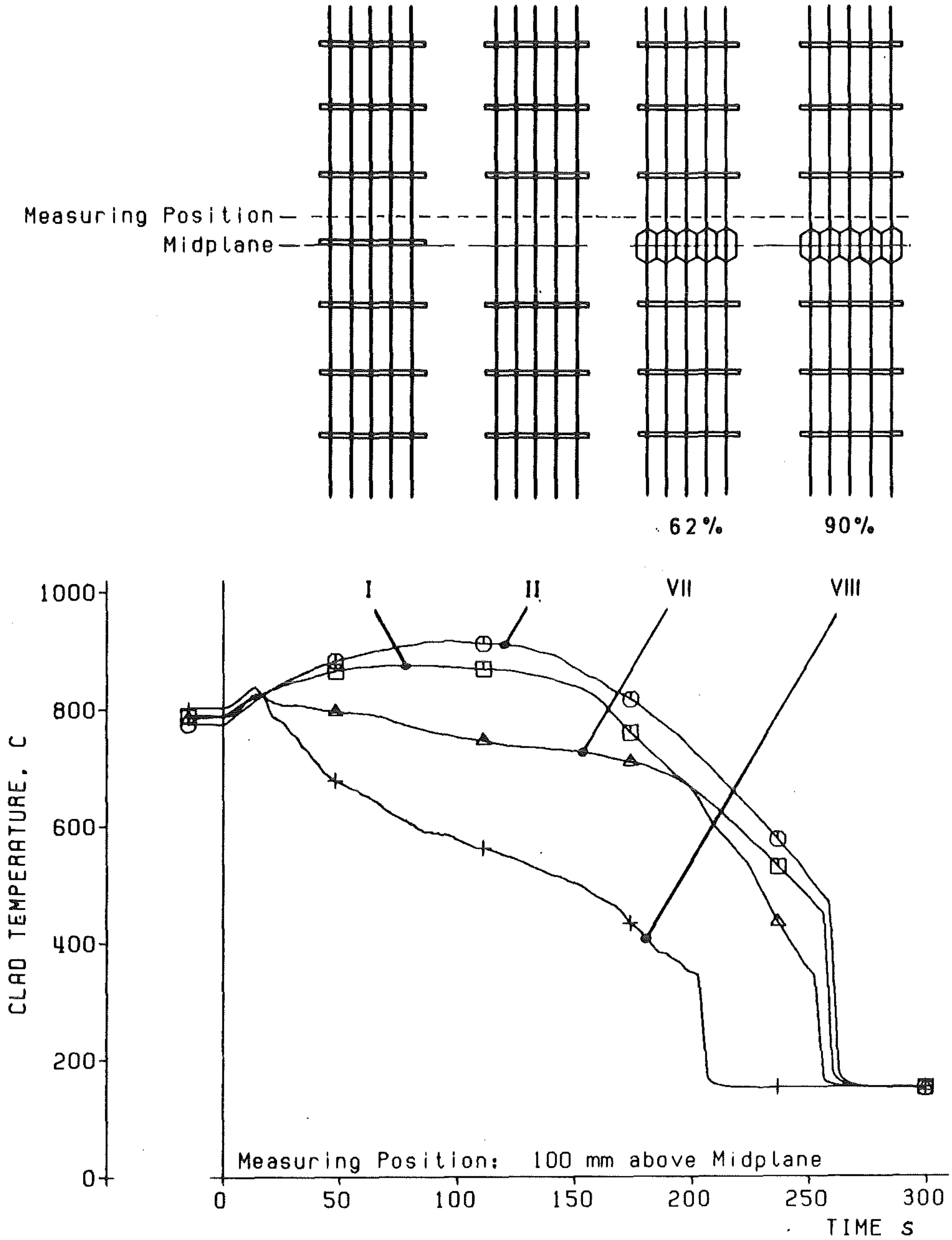
Fig. 9 INFLUENCE OF GRID SPACERS AND BLOCKAGES ON CLADDING TEMPERATURES FOR IDENTICAL COOLANT MASS FLUX IN ALL SUBCHANNELS

2.3 Effect of Two Subsequent Blockages

From Zircaloy cladding behavior tests in rod bundles during refill and reflood, it is known that conglomeration of ballooning may occur at two different axial levels separated by a grid spacer position /16, 17/. Therefore, the 62 % and the 90 % partial blockages, separated by a 300 mm axial distance, were mounted in the same 3 x 3 rod cluster.

The main question to be answered from this test series was, whether the main coolant flow would bypass both blockages and cause a hot region between the two blockages. Figure 10 shows the axial temperature profile obtained with a flooding velocity of 2 cm/s and a system pressure of 4 bar.

The profile is "snapshot" 150 s after start of flooding, i.e. at the time of peak cladding temperature. Along the flow direction, the sleeve temperature decreases within the 90 % blockage as known from the test data shown before. The cladding temperatures in the bypass do not decrease inspite of the locally increased mass flux. Such an effect is observed only for higher flooding velocities. A hot region occurs downstream of the 90 % blockage and upstream of the grid spacer. High cooling enhancement can be detected upstream of the 62 % blockage where a hot region is expected. However, the throttling effects of the grid spacer as well as those of the following 62 % partial blockage are equalizing the flow over the cross section of this part of the bundle. Enhanced droplet dispersion in the turbulent flow then leads to improved cooling. The sleeve temperatures in the 62 % partial blockage and the cladding temperatures downstream of it are lower than the cladding temperatures in the bypass as known from the test data shown before. However, a second region of higher cladding temperatures can be observed about 200 mm downstream of the upper blockage. The cladding temperature profiles shown have developed during the mist flow regime. The water content in the superheated steam has been consumed by increased evaporation for the flow passing through the blockages. Therefore, downstream of the upper blockage the heat produced in the rods has to be removed by mist flow of lower water content than that in the bypass rod cluster.



Flooding Parameters: V = 3.8 cm/s, P = 4.1 bar

□ Test Series I, with Grid Spacer at Midplane
○ Test Series II, without Grid Spacer at Midplane
△ Test Series VII, all Subchannels 62% Blocked at Midplane
+ Test Series VIII, all Subchannels 90% Blocked at Midplane

Fig. 10 AXIAL TEMPERATURE PROFILES IN THE BLOCKED AND THE BYPASS ROD CLUSTERS OF A 5x5 ROD BUNDLE WITH TWO SUBSEQUENT BLOCKAGES

2.4 Heat Transfer Analysis

With respect to cladding deformation during reflood, the heat transfer conditions at the very beginning of reflood are most important. Therefore, this brief discussion of heat transfer analysis is concentrated on the mist flow regime conditions. For flooding velocities as low as 2 cm/s, water is carried through the whole bundle length. About 100 s after start of reflood, already a significant amount of water had been measured after separation of the water from superheated steam beyond the bundle exit /11/. The volume fraction of water in the two-phase mixture is significantly lower than one percent. However, its cooling effect is of great importance /24, 25/.

The droplets entrained by the steam are intercepted by flow obstacles such as grid spacers, ballooned claddings, burst lips etc., and dispersed or reentrained /26/. The larger surface to volume ratio for decreasing size of droplets favors enhanced evaporation. Moreover, there may be such other phenomena as in the wake of partial blockages: Large droplets falling back due to gravity into regions of reduced steam flow, their dispersion approaching hot surfaces, and steam jets leaving the constricted subchannels. It is difficult to model these complex mechanisms. They have to be selected depending on their cooling effectiveness. Important differences occur downstream of blockages in analyzing single phase or two-phase flow heat transfer conditions. Besides the efforts to model three-dimensional two-phase flow using a fully three-field representation /13/, some empirical methods are proposed. One of such methods is the use of single-phase flow diversion analysis around a blockage. Adjusting the heat transfer for two-phase flow conditions, empirical values for the locally disturbing effects of the blockage are used as multipliers /14, 15/. However, reflood heat transfer analysis for undisturbed core geometry still contains some uncertainties covered by conservative assumptions. Therefore, there is not a generally accepted model for suitable calculation of cladding temperatures in and around blockages.

Heat transfer analyses of experimental data show the following trends:

1. The change of the heat transfer downstream of a blockage compared with unblocked bundle conditions is time dependent.
2. The ratio of the heat transfer coefficient (HTC) blocked to HTC unblocked decreases during flooding.
3. For the 90 % partial blockage shown before, HTC blocked to HTC unblocked decreases from about 1.0 at the very beginning of flooding to about 0.65 short time before quenching of the corresponding region.
4. For the 62 % partial blockage the initial value starts above 1.0 depending on the flooding conditions and decreases down to or below 1.0.

Figure 11 shows these trends qualitatively versus normalized quench time $t' = t/t_Q$. The analysis of the experimental data was carried out for the grid spacer effect /24, 25/.

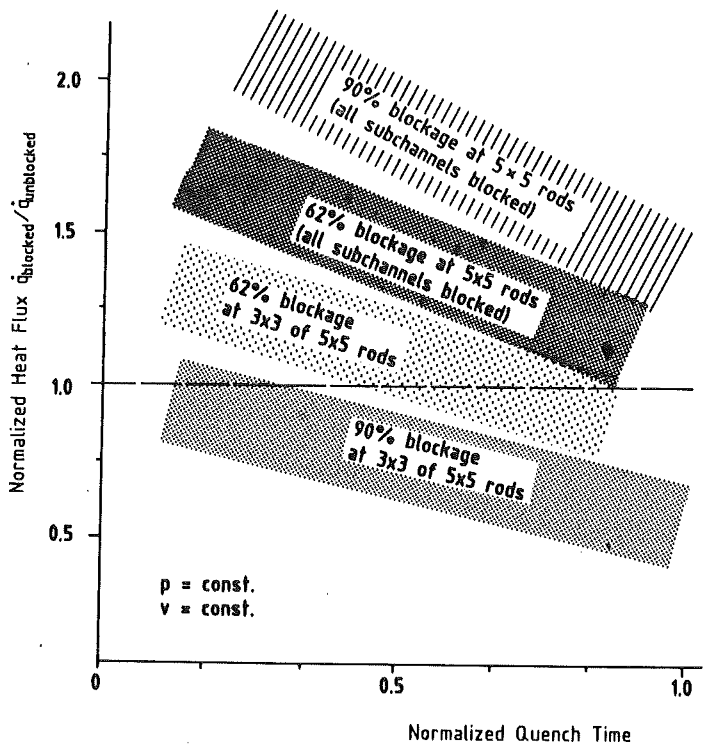


Fig. 11 NORMALIZED HEAT FLUX BLOCKED/UNBLOCKED DOWNSTREAM OF BLOCKAGES IN A 5x5 ROD BUNDLE OF PWR DIMENSIONS DURING PRECURSORY COOLING OF REFLOOD

The use of the heat transfer coefficient based on either the saturation temperature or the fluid temperature is not always adequate to describe the heat transfer conditions during mist flow regimes. Since, especially for low flooding velocities the steam is highly superheated and the droplets are at the saturation temperature, the description of the heat removal from the rod claddings to the coolant is difficult. Radiation as well as convection from the claddings to the superheated steam as well as to the liquid phase occurs. Furthermore, there is heat exchange between the two phases of the coolant in the bulk as well as in the boundary layer penetrated by droplets. Therefore, the surface heat flux, being the result of the different complex mechanisms, has been chosen for presentation of some selected experimental results. Figure 12 shows rod surface heat flux transients at different axial locations from the 62 % partial blockage test performed with v = 2.2 cm/s, p = 4 bar (compare Fig. 6). The data represent the conditions in the blocked rod cluster. Figure 13 shows the heat flux transients about the 62 % blockage, constricting all subchannels, evaluated from a test recently performed with v = 2.2 cm/s and p = 4 bar (compare Fig. 7).

To answer the open questions about local cladding temperatures due to pellet eccentricity and fuel slumping, mentioned in Section 2.1, only a small amount of experimental data gives indications about the definitive consequences. However, the effects can be estimated. Taking into account e.g. the boundary conditions of a 62 % partial blockage discussed with the aid of the temperature transients shown in Fig. 6, the following picture can be deduced:

If one of the rods simulating pellet columns would touch a sleeve simulating a ballooned cladding, the upper temperature limit of the sleeve would be the rod surface temperature shown. For a fuel rod cladding, ballooned up to the extent of the shape of the 62 % blockage sleeve, the upper temperature limit at a single spot of the balloon would be on the same order of magnitude under the condition that the cladding touches a pellet. This limited hot spot would show temperatures of about 200 K above the average balloon temperature. This temperature level will not be exceeded and it will remain localized whether or not the cladding bursts. If the cladding did touch a pellet before deformation up to this extent, due to internal overpressure, it would burst earlier with the consequence of less ballooning, i.e. less coolant channel constriction.

For the case of slumping of broken fuel pellets into the balloon before burst of the cladding, the consequences seem to be more severe. The results of the FR2 in-pile tests /19, 38/ showed fuel relocation which occurred at or after burst. Assuming relocation before burst, the consequences can be estimated roughly for the conditions of the 62 % partial blockage mentioned above: If the volume surrounded by the cladding increases by a factor of about 1.8, i.e. for the size of the balloon mentioned, and the volume fraction of the broken pellets is in the range of 60 % of the original pellet density, - typical value for rubble in cylinders -, the volumetric power generation increases about 8 % for the center of the balloon and the center of the blockage, respectively. For the tests discussed in this paper a 120 % of ANS Standard decay heat power has been applied which covers the increase estimated. However, the gap coefficient, which was about 0.03 W/(cm K) for the sleeve test conditions, could increase by a factor of 10 or more due to closer contact of pellet fragments with the cladding (compare Fig. 14). Thus, the most pessimistic assumption leads to temperatures of the whole balloon close to the fuel temperature which amounts max. 200 K higher. This estimation can be deduced analyzing the conditions for a sleeve blockage. The heater rod temperature underneath the sleeve approximately corresponds to the fuel surface temperature underneath a balloon (see Figs. 4, 5 and 6). Compared with the cladding temperature measured in the bypass this is roughly the same transient for the early portion of the reflood phase. Reasons for this behavior are the increased surface of a blockage exposed to the coolant as well as increased droplet cooling which is increasingly involved in disturbed coolant channel geometries.

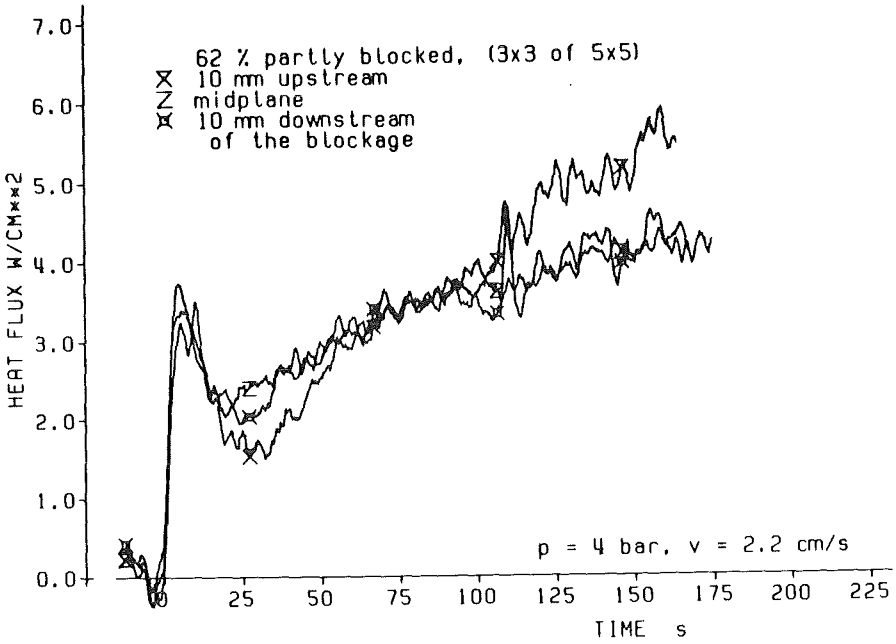


Fig. 12 ROD SURFACE HEAT FLUX UPSTREAM, UNDERNEATH, AND DOWNSTREAM OF A 62 % SLEEVE BLOCKAGE AT 3x3 RODS IN THE 5x5 ROD BUNDLE

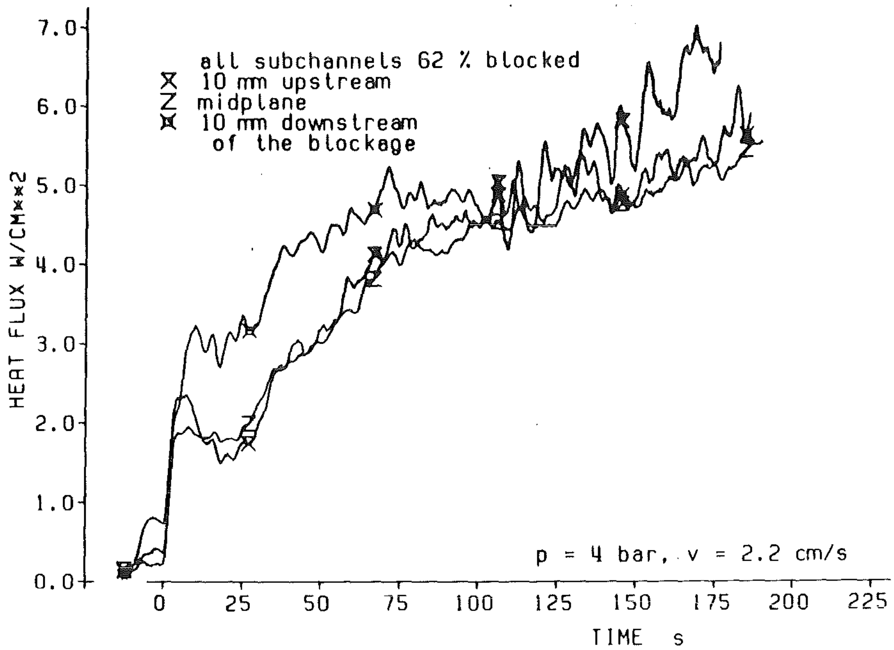
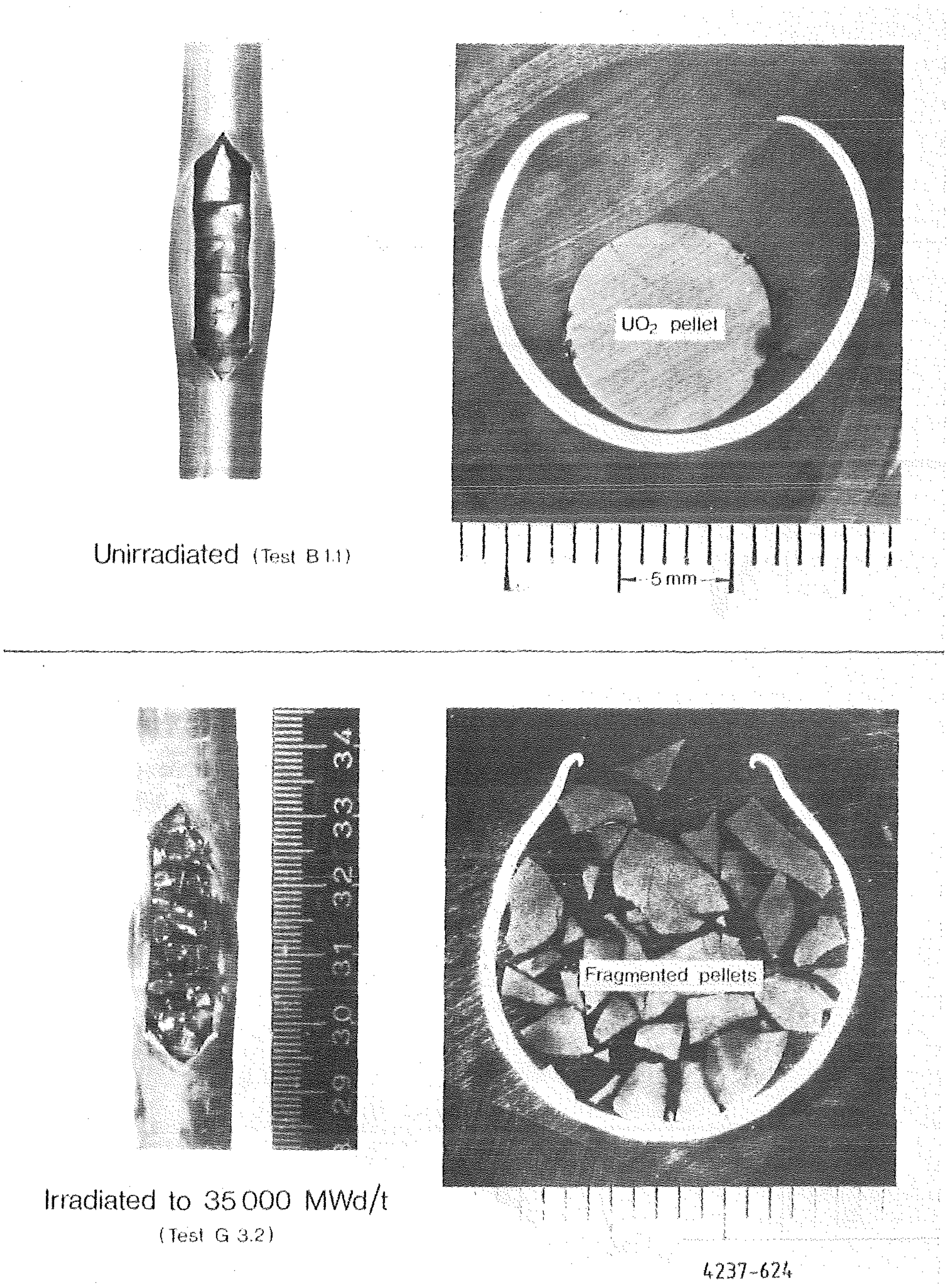


Fig. 13 ROD SURFACE HEAT FLUX UPSTREAM, UNDERNEATH, AND DOWNSTREAM OF A 62 % SLEEVE BLOCKAGE AT 5x5 RODS IN THE 5x5 ROD BUNDLE



From: E.H. Karb et al.: "LWR Fuel Rod Behavior, FR2 In-pile-Tests" /19/

Fig. 14 FUEL PELLETS IN UNIRRADIATED TEST RODS (a) USUALLY REMAINED INTACT DURING TRANSIENT TEST; PELLETS IN IRRADIATED RODS (b), ALREADY CRACKED DURING PREVIOUS IRRADIATION, WERE FOUND FRAGMENTED AFTER THE TRANSIENT TEST IN SECTION WITH MAJOR DEFORMATION

3. LONG-TERM COOLING OF A CORE AFTER A LOCA WITH SEVERELY DEFORMED FUEL ROD CLADDINGS

The second heat-up phase of the core during a LOCA transient, i.e. during the partial or complete dryout of the reactor vessel after blowdown, has to be interrupted refilling the vessel and reflooding the core. If during this period a situation had led to severe clad deformations in the center of the core, three questions arise for the quasi steady state long-term heat removal:

1. Which portion of the core is more critical for reduced long-term cooling water injection, the upper end or the blocked center?
2. Can this blocked portion in the center of the core be or kept cooled down?
3. Under which conditions could it dry out again during long-term cooling?

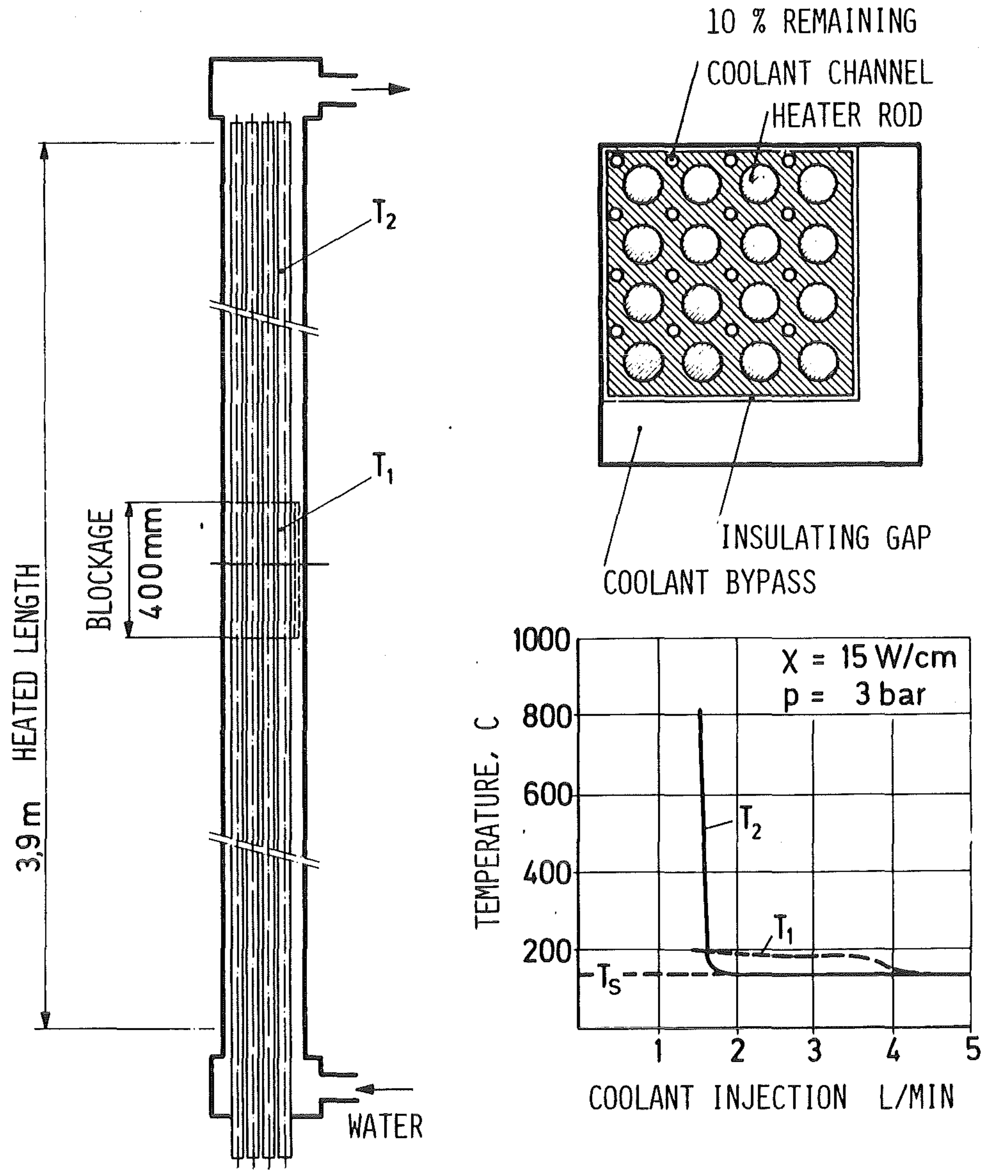
To answer these questions steady state long-term cooling tests were performed with a bundle of 16 electrically heated rods of PWR dimensions using the FEBA facility /30/. In these investigations the blockage simulated the most pessimistic coolant channel constriction of 90 % over a length of 0.4 m at all subchannels in the midplane of the 16 rod cluster with large bypass. Inspite of the limited number of rods, the radial extension of the blockage was simulated to be infinite due to the radial insulation of the blockage device. The heat generated in the rods had to be removed in radial directions to the surface of the individual subchannels and to the coolant, respectively. The subchannels of 90 % blockage ratio were represented by cylindrical holes of 3 mm diameter. Thus, the surface exposed to the coolant was severely reduced.

The initiation of the tests was carried out for both modes, quenching down the whole array from an initial temperature level of about 700 °C (flooding test), and boiling out the bundle (boiling test), investigating the dry out locations during the steady state long-term heat removal. The results obtained for both test modes were the same /30/.

The following parameters have been varied and kept constant during each test: Rod power (10 through 25 W/cm), injection rate (1.5 through 6.0 l/min, i.e. 1.4 through 5.6 cm/s for the rising water level in the cold test section), and the system pressure (3 and 5 bar). Typical results are shown in Fig. 15. For the lowest injection rate and a maximum rod power of 15 W/cm in the blockage region surface temperatures increased above 600 °C at the upper end of the bundle far downstream of the blockage. For a maximum rod power of 25 W/cm, unrealistic for long-term cooling conditions, surface temperatures increased above 600 °C within the blockage while at the upper end of the bundle they remained below 600 °C.

In the investigation described the following conclusions were drawn:

1. For realistic power levels and flow conditions an extreme 90 % blockage in the center of a PWR core does not overheat during long-term heat removal.
2. With decreasing injection rates, dryout starts at the upper end of the rods rather than in the blocked area.
3. It seems to be sufficient to keep the core covered with water maintaining a corresponding minimum water injection rate of about 1.5 cm/s whether or not severe blockages at the bundle midplane are present.



From: G. Hofmann, W. Baumann: "Long-Term Coolability of a Partially Blocked Core, Experimental and Theoretical Results" /30/

Fig. 15 DRYOUT CONDITIONS AT CENTRAL 90 % BLOCKAGE AND UPPER BUNDLE END DEPENDING ON ROD POWER AND INJECTION RATE

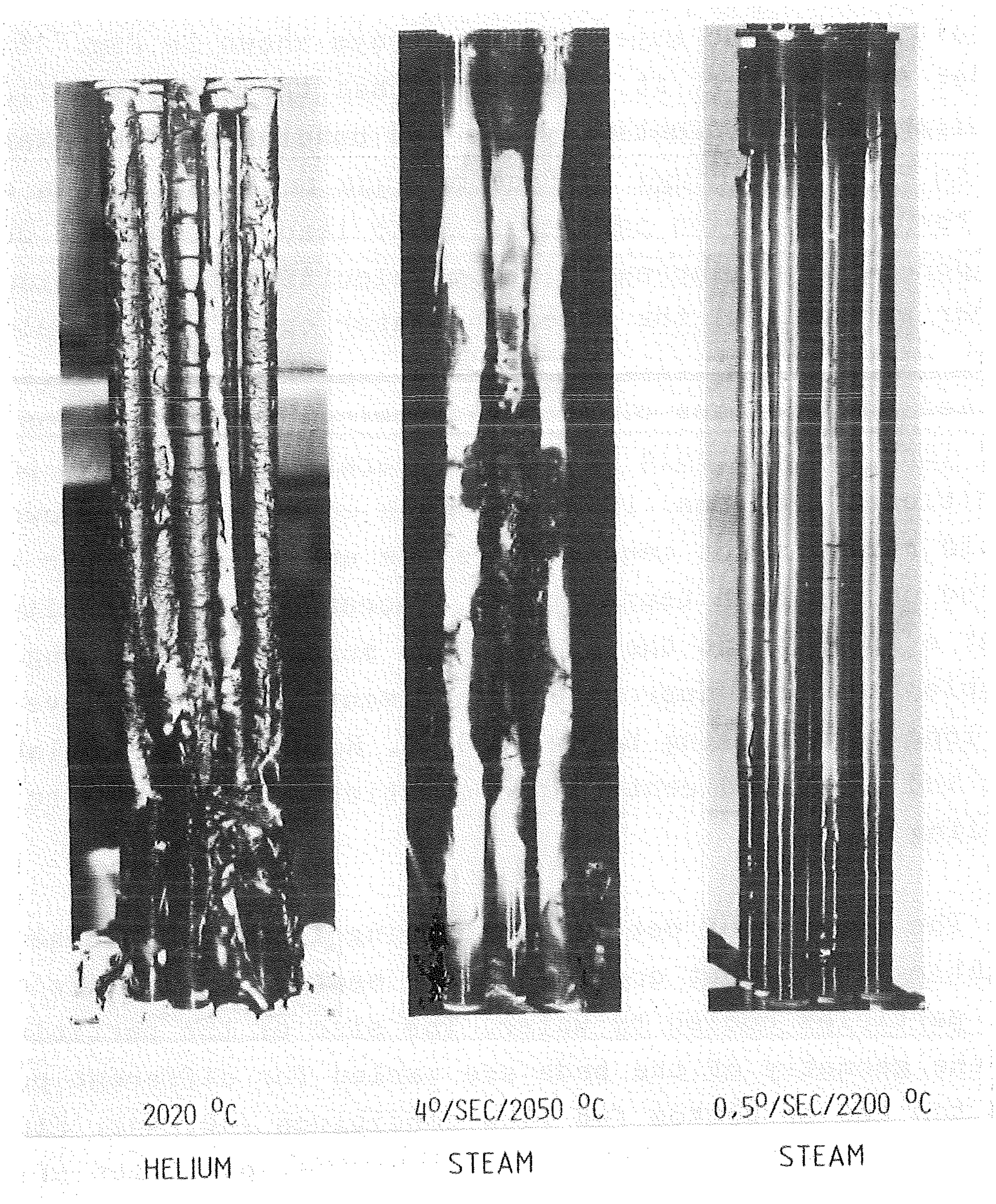
4. BEGINNING OF SEVERE FUEL DAMAGE, BUNDLE AND CORE DISRUPTION

For the case that the second heatup phase of the core during a LOCA is not intercepted fast enough, maximum temperatures occur in the center of the core first. At the onset of reflood the fuel rods might not only have been damaged by ballooning but also by enhanced oxidation and/or partial melting of the cladding. Depending on the temperature transient as well as on the temperature level of the rods and the conditions of the surrounding atmosphere, complex interactions take place between the different components of the core. Assuming a roughly intact rod bundle geometry at the onset of reflood, significantly different configurations can be expected as shown in Fig. 16. Out-of-pile experiments without cooling led to the bundle deformations presented in this figure /28/. Further experiments using single rods and bundles /29/ serve for preparation of severe fuel damage experiments including reflood.

Up to now, experimental data for the reflood cooling conditions in such geometries are not available. Therefore, this question can be discussed only briefly.

With respect to the content of Section 2.1 concerning emergency core cooling for an initial temperature level of the core below 1000 ^{o}C, the initial conditions for reflooding severe fuel damage configurations are different e.g. by the following reasons:

1. The temperature possibly has reached a level favoring further chemical reactions.
2. The steep temperature gradients from the entering coolant to the materials to be cooled are favoring mechanical disruption and formation of debris beds.
3. The mechanical stability is lost more or less for severely damaged portions, since the claddings became brittle or were molten down. Grid spacers in that area failed even earlier. However, the coolability of the core without significant disruption of the damaged portion depends on how far it is cooled down before debris beds are formed.

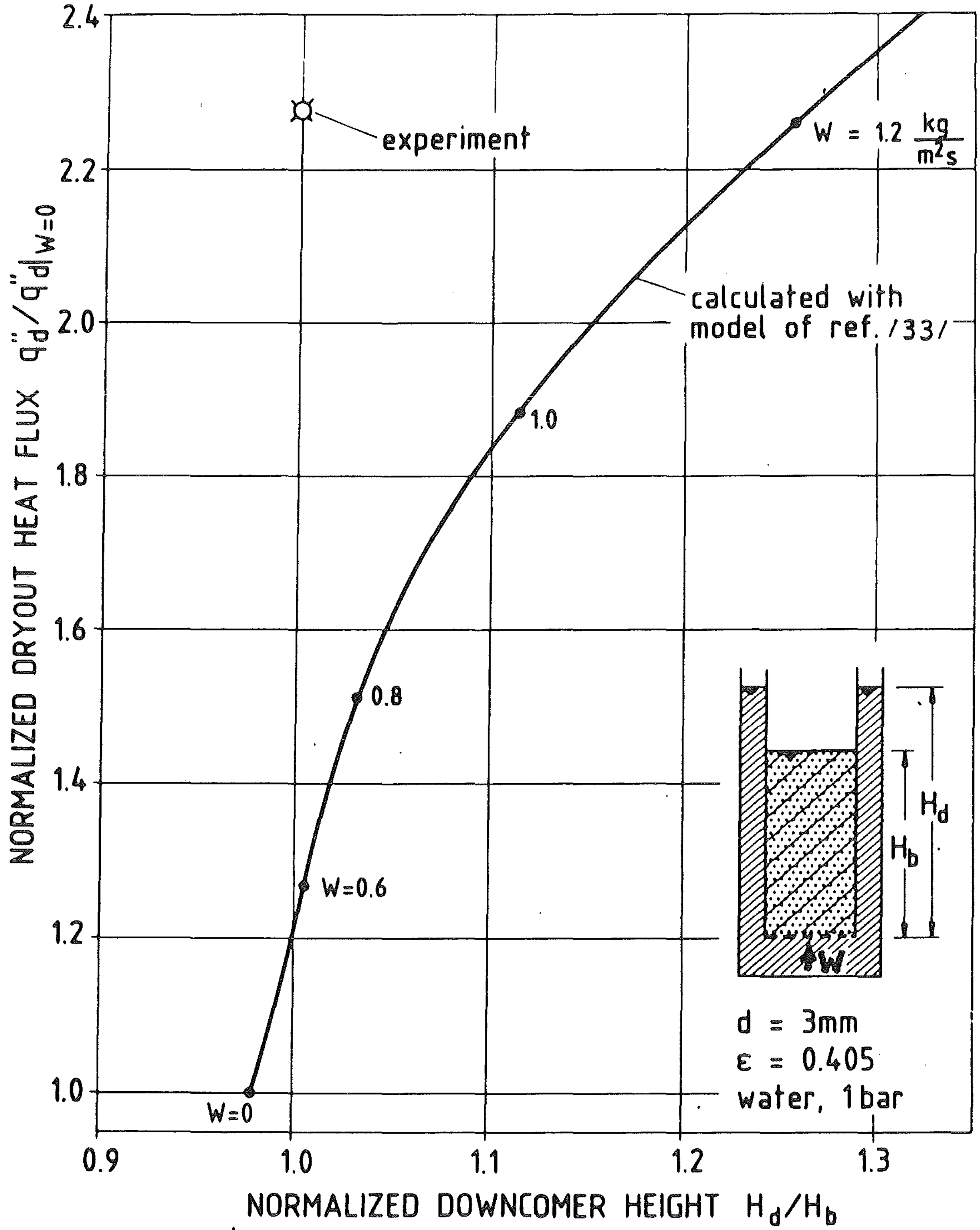


From: S. Hagen, H. Malauschek: "Bundle Experiments on the Meltdown Behavior of PWR Fuel Rods" /28/

Fig. 16 KFK-BUNDLE EXPERIMENTS ON THE MELTDOWN BEHAVIOR OF PWR FUEL RODS WITHOUT REFLOOD COOLING

Returning to the sketch shown in Fig. 1 and to the cooling conditions discussed with the aid of e.g. Fig. 10 the following conditions can be expected for arrays shown in Fig. 16: At the onset of flooding, it is assumed that the dispersed flow will be cooling fairly effectively the rod bundles with blockage ratios up to 90 %. The rough surface of nude pellets, broken claddings, molten and refrozen particles would lead to enhanced droplet evaporation and increased steam velocities for the middle and the upper portions of the core. The water carry over - being a lost heat sink /11/ - would be reduced significantly. It can be expected that regions of te core showing blockage ratios of 60 % and less are coolable without temperature excursions posing additional problems. If most of the core is cooled down, there could remain a hot center where the entrance of water is inhibited due to high temperatures, increased power density and violent evaporaion at the steam/water surface surrounding. This problem has been approached experimentally in the investigation of long-term cooling below 1000 °C. However, the coolability of a hot and disrupted center is a question more related to debris bed cooling.

The problem of debris bed cooling is being assessed investigating the dryout conditions for heated beds mainly of spherical particles cooled by water. The size of the particles as well as the geometry of the beds are varied for different power levels and top fed or bottom fed flow conditions /40, 41/. Figure 17 shows an example of a bed using spherical particles of 3 mm diameter cooled by downcomer driven bottom injection of water. The experimental data, first results of a program recently started, indicate significantly higher dry out heat flux than data calculated. This discrepancy demonstrates the necessity of investigations being under way in this domain.



From: G. Hofmann: "On the Location and Mechanisms of Dryout in Top-Fed and Bottom-Fed Particulate Beds" /3/

Fig. 17 INCREASED DRYOUT HEAT FLUX IN A BED WITH DOWNCOMER DRIVEN BOTTOM INJECTION

5. CONCLUSIONS

- Heat transfer conditions in rod bundles with severe clad deformations are partially understood.
- Experimental data exist from out-of-pile tests for simulated core configurations with highly ballooned fuel rod claddings (temperature range below 1000 °C).
- The coolability of rod clusters blocked up to 90 % during the transient reflood phase is not a severe problem even for flooding velocities as low as 2 cm/s.
- In-pile tests for confirmation of the results obtained out-of-pile are under way /20/.
- The analytical and the semi-empirical methods for description of the cooling mechanisms seem to need improvement and have to be verified using experimental data to reduce conservative assumptions.
- The main interesting flow regime during reflood is the mist flow regime characterized by low water content and high steam superheat.
- Long-term coolability of rod bundles with severely deformed rod claddings during the quasi-steady phase after a LOCA is no problem for blockage ratios up to 90 % and water injection rates corresponding to flooding velocities above 1.5 cm/s.
- The cooling conditions in fuel rod clusters with oxidized, broken and partly molten claddings and fuel are not well known (temperature ranges above 1200 °C).
- Analytical work and experiments (out-of-pile, in-pile) are under way to answer open questions.

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