7TH ANNUAL CONFERENCE ON NUCLEAR SAFETY RESEARCH

# Reactor Safety Research in the Federal Republic of Germany

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## 1. Introduction

Considering the dense population in the Federal Republic of Germany the growing demand of nuclear power stations makes reactor safety research even more necessary than in other less dense populated countries. This was one of the reasons why the Minister for Research and Technology announced a research program for light water reactors with a budget of about 200 Mio German Marks for the years 1972 until 1976. A self evident classification of such a research program can be deduced from the requirements:

1. to avoid accidents,
2. to master the effects of a possible accident without damage or harm to the neighbourhood of the power station and
3. to remove after the accident the consequences as soon as possible.

Regarding the todays' knowledge and know-how in reactor safety this classification is too general and therefore to straiten the German reactor safety program the following classification was chosen.

1. Closing of gaps in the knowledge about reasons and course of accidents and improving the understanding of hypothetical accidents.
2. Improvement of the safety by operational means.
3. Closing of gaps in our knowledge which up to now were bridged by conservative assumptions in the licensing procedure.
4. Improvement of components.

The research program which up to now consists of more than 100 different research activities is planed and discussed in several advisery committees, which report ot the Ministry of Research and Technology. At time there exist the following advisery committees:

Blowdown in the containment,
Emergency core-cooling,
Core-melting,
Pressure vessel failure,
Influences from outside,
Quality securing,
Material and strength,
Removing of accident effects.

Furtheron the Minist ry for Technology and Research has installed a special department at the German Institute for Reactor Safety to attend this research program. This institution publishes every three months short progress reports /1/ of all research activities in nuclear safety.

## 2. Research Program

For a short discussion of the research program let us divide all activities in the three categories:

1. avoiding of accidents, (Tab. 1)
2. mastering the effects of accidents, (Tab. 2)
3. removing the effects of accidents. (Tab. 3)

### 2.1 Avoiding of Accidents

Research work to avoid accidents has to start from a very diligent system analysis about reasons and possibilities of accidents. This also includes design measures as well as material analysis for reactor components. A continuous supervision during reactor operation by means of repeating quality tests with respect to crack and failure behaviour helps to reduce the possibilities of a mechanical failure of the reactor components enormously. But accidents also can be created by wrong actions of the operator, therefore also the training of operators as well as the analysis of the human functions are of great importance. Finally one also has to mention measures to avoid hypothetical accidents and therefore also research activities are underway for burst protection of pressure vessels and components.

Activities dealing with the nuclear dynamics and the thermohydraulics in the core are underway since many years in the "Laboratorium für Reaktorregelung und Anlagensicherheit" (LRA = Laboratory for Reactor Control and Safety). The main effort is given to some specific questions like the analysis of reactivity excursions, taking in account the thermo-hydrodynamic feed-back for boiling water reactors.

Special effort also is given to a detailed system analysis. Here mainly the influence of special parts of the reactor protection system and of the controlling devices on the availability of the power station is researched.

The design and layout of reactor components have doubtless a special meaning in avoiding accidents. Within the research program in first order therefore activities should be mentioned, which were underway in the laboratory for material testing in the Technical University Stuttgart and which deal with a special program for the pressure vessel steel "22 Nickel-Molibdanium-Chromium 37". These activities mainly clarified the questions in connexion with welding this material and worked out informations about possible critical behaviour in connexion with burst security of reactor pressure vessels.

Besides this the whole research program at this laboratory of the Technical University Stuttgart comprises a much broader frame. It starts from researching the possible influences which may diminish the quality of the pressure vessel and which may perhaps not, or not completely, be detected by the usual quality testing devices.

All activities of this program aim in a reliable prediction of the failure probability of pressure vessels and in a diminuation of this probability by useful quality improving measures. So the whole program deals with tests regarding the burst security, with deliberations to optimize the pressure vessel technique and with the development of new pressure vessel design.

To improve the quality it is not enough just to research the physical behaviour of the material during a crack growing. The activities have to be completed by design work supported on a theoretical and experimental analysis. This design work is mainly concentrated on the flanges and connecting pieces of pressure vessels.

The quality of a reactor pressure vessel and other pressure imposed components is based on the quality of the testing procedures and testing devices during manufacturing and during operation. Here the development of none destructive testing methods in thick walled vessels is very important. Beneath the X-ray testing the ultrasonics method for repeating tests has the highest grade of development. Experimental and theoretical work to use this method in practical pressure vessel testing is done since years in a joint research program where the companies M.A.N., Krautkrämer, KWU and the National Institution of Material Testing in Berlin take part. Aim of this development work is to test not only the pressure vessel itself but also all other primary components. The automatically working pick up and measuring devices were already used in several nuclear power stations. They allow the testing of the pressure vessel as well from outside as from inside.

Beneath the ultrasonics technique there are some other measuring methods under development. One has perhaps to mention the noise analysis during crackgrowing, the eddy current method and the utilization of stochastic signals coming from the neutron flux or the pressure and temperature fluctuations of the primary coolant. It is the intention to develop the noise analysis to a continuously acting warning system for pressure vessels. Furtheron it is intended to use this method for location of flaws in connection with a repeated pressure test. By using statistical methods one also gains knowledge about the local and temporal dynamic behaviour of the reactor and gets hints of dangerous vibrations of the core structure. These vibrations can induce stochastic changes of temperature and density in the cooling fluid via reactivity effects.

Each technical system even with extensive automatization finally is controlled by human beings. Therefore the training of reactor operators is a main subject within the research efforts concerned with reactor safety. The training program was started with a study informing about the todays status of operator instruction and it worked out in detail the future traning program. Finally there will be set up theoretical function-schemes of the human behaviour in nuclear power stations.

By means of a careful and extensive testing for quality securing it is doubtless guaranteed that the existence or the growing of a critical crack and with it the failure of the pressure vessel can be eliminated with a certainitylike probability. Locating nuclear power stations in the vicinity of cities makes it necessary to take in account also smallest risks which due to their extreme small probability are usually neglected in the legal and technical practice. Under these deliberations in our safety research program activities were started to design and to layout burst protection devices for reactor pressure vessels and other components of the primary sytem. After a careful study of different constructional possibilities and after an extensive development the German Company KWU has forwarded a technically thoughtover design for a burst protection device.

To develop and to test this device several research work has to be done which mainly deals with the evaluation of the critical loading distribution, the strain of tension-components and the thermal insulation properties of concrete. Another problem is the inspection of the pressure vessel if the burst protection is around it.

## 2.2 Mastering of Accident Effects

An accident may arise by a failure in the system, that is by an reactor internal event but also by external influences like earthquake, airplane crash or explosions in a gascloud. The maximum credible accident caused by a failure in the system is the break of a large pipe filled with primary coolant. If this happens it has to be guaranteed that no unpermissible burden arises for the near and far neighbourhood of the nuclear power station. The containment serves to hold back the radioactiv products and its integrity has always to be guaranteed.

As known the core develops after shutdown still heat which has to be carried away for a long time at allowable temperatures of the fuel rods. Therefore a series of reactor safety problems arises within the emergency core cooling. We shall deal with this problem in the next chapter more in detail. If there is unsufficient emergency cooling this could lead to a melting of the fuel or of the core structure. But this would mean that the extreme unprobable case occurs that all emergency cooling systems which are available in multiple redundance do not work. If one intends to master also extreme hypothetical and unprobable accidents and their effects one also has to do - at least for a better understanding of this problem - some research work in core melting.

Research work in blowdown-problems is under way since many years at the Battelle-Institute in Frankfurt. At the time being they mainly are clarifying the pressure behaviour in the very first seconds of the blowdown, mainly the critical mass flow rates during this time and the stress on the core structure due to pressure differences.

In an international cooperation blowdown tests were carried out at the nuclear power station Marviken, which is out of operation. The main aim of this activities was to get a substantial experimental basis for comparison with existing computer codes which for example were developed by the LRA at Garching. Furtheron the behaviour of iodine release and the mechanical stresses at different reactor components were tested.

An incident in the German nuclear power station Würgassen at which after opening of a valve the bottom of the pressure-subpression chamber was damaged by pulsating pressure load gave rise to detailed and extensive research activities dealing with the condensation behaviour of steam blown into water. After first tests with small models the experiments were carried out in a big scale in a conventional power station at Mannheim. The experiences gained about the dynamic behaviour during the pressure subpression period led to a new design of the steam pipe nozzels which now guarantees the integrity of the bottom of the pressure-subpression chamber.

As known the containment includes several seperate rooms which are connected with each other by window like openings. If one of these openings is not big enough, a too high pressure could be built up during the blowdown, which could distroy a wall and so endanger the integrity of the containment. To study the pressure and flow behaviour in these chambers a test containment as shown in fig. 1 was constructed at the Battelle-laboratories in Frankfurt. With this arrangement all important parameters, like pressure temperatures and mechanical stresses in the different rooms of a multiple subdivided containment, will be measured. The experimental data will be compared with results given by a computer code which was evaluated by the LRA Garching. By this way this computer code will be so improved that it is possible to predict with high accuracy the behaviour of the containment expected under orginal conditions.

Before discussing the problems of emergency core cooling and fuel rod failure in detail just a few comments should be given about the activities in the core melting program. Experimentally an integral big core melting test seems impossible to be carried out under realistic conditions. In addition there would be no possibility for extrapolation looking for different parametric influences and so the knowledge gained from it would be very limited. Considering this one has to regard experiments now started in our country which mainly have the aim to get a clear knowledge about the influence of different parameters if there would be core melting due to a break down of all emergency core cooling systems. This tests mainly deal with

the mutual influences between cladding and pellets at different temperatures,

the behaviour of a $UO_2$ fuel rod when melting starts,

the course of the melt down and the recrystallization of the molten fuel at cool parts of the rod.

These tests are carried out at the National Nuclear Research Center Karlsruhe and are started with one fuel rod under steam atmosphere. They are then going on with a bundle of 4 rods.

Theoretical and experimental work dealing with the thermohydraulic behaviour of the molten core are done at the Institut für Verfahrenstechnik of the Technical University Hannover. The results worked out up to now give information about the heat transfer coefficient between the molten core and its border, i.e. the pressure vessel or a core catcher. An extensive and detailed computer code allows to predict whether and how long the molten fuel can be kept in the pressure vessel by means of a suitable cooling of the pressure vessel bottom and how a core catcher for such a pure hypothetical accident has to be laid out and designed. For the experiments an optical measuring technique, the holographic interferometry was used. Fig. 2 shows a comparison of measured and theoretically by the computer code determined isotherms in the molten core inside the pressure vessel. Fig. 3 gives the local heat transfer coefficient around the surface of the pressure vessel.

The metallurgical and chemical interactions between the molten core and the pressure vessel wall are influenced by the thermodynamic properties as well as by the hydraulic behaviour of the molten fuel. Research work about the metallurgical problem is under way at the KWU in Erlangen in which in a small vessel molten fuel of different composition is tested with respect to its metallurgical interaction with the pressure vessel steel. Another problem is the pressure energy, which arises from steam explosions due to interaction between liquid fuel or liquid structure material and water, for which tests are started at the Euratom Research Center in Ispra. Finally one has to mention measurements about the radioactive release during the melt-down which are carried out at the Nuclear Research Center Karlsruhe and which should compliment the data already given in the literature.

Shortly it should be mentioned that there are activities under way which deal with aircrash-problems and with the possible influences of an ignition of an explosive gas cloud. Furtheron studies are done to improve the design of the nuclear power stations against earthquake. There are some deliberations to use perhaps the superheated steam reactor (HDR) in Kahl as an test-object which is out of operation.

## 2.3 Removal of Accident-consequences

Mastering and removing accident-consequences is mainly a radiological problem. The biggest part of radiological research within the nuclear safety program is under way in the National Laboratory Karlsruhe. The radiological burden of the environs of the nuclear power station is mainly determined by the efficiency of the filters. For developing new filter concepts and for doing measurements with them a big testing device is available in the National Research Center Karlsruhe.

Nuclear safety deliberations showed that by developing recycling filters the endangering risk for the environs of the power station can be extremely deminuated. Therefore there are strong efforts to develop recycling filters for nuclear power stations but also for the air outlet in fuel processing stations. Furthermore there is a seperation process under development for the radioactive inert gases coming out from the processing plants. The aim of this research efforts is to get a "minimum release-concept" for the accident case and naturally also during normal operation.

Finally there has to be mentioned work which deals with removing the accident consequences in the power station itself. These activities - also under work in the National Laboratory Karlsruhe - can be subdivided in the development of remote-controlled manipulators and in insulating and removing contamination.

For the decontamination of radioactive rooms there are different possibilities. At time the National Laboratory in Karlsruhe develops a new process which works with foam controlled non-ionogen tensid mixtures which are tested at different surfaces on their decontamination action. This process needs acids to form complex groups. Such acids are formic acid, lactic acid, oxalic acid and tartaric acid. It could be shown that with this method good results can be achieved on PVC-flooring and also on stainless steel surface. Within these problems one also has to mention activities for treating large amounts of radioactive liquids at the place of the accident and also treating solid waste which cannot be decontaminated. Finally there is the problem of rem ng and storing tritium.

## 3. Emergency Core Cooling

Due to its significance emergency core cooling research plays an important role in the reactor safety program of the Federal Republic of Germany. Experimental and theoretical activities dealing with the thermohydraulic behaviour in the core during the blowdown and the refilling period are under way since many years all over the world. Keeping in mind the complex character of the emergency cooling problems which exclude a pure theoretical treatment it seems to be obvious to carry on the experiments in that way that one imitates the primary system of a nuclear reactor including its emergency core cooling devices as good as possible and tries to make the tests like in the real accident. Similar planned tests were carried out years ago in the U.S.A. and showed mainly in the first testing periods very spectacular results. One may remember, for example, the so called loft-semi-scale tests in Idaho.

Thinking over such tests the question immediately arises how the testing conditions in a small scaled experiment - in view of the very complex hydrodynamic and thermodynamic conditions - are really reactor-similar and how they can be extrapolated to the original conditions given in the nuclear power station. Already a very first rough deliberation about the problem - there are unsteady two-phase-flow, heat storage, fast changing heat transfer conditions and strong geometrical dependent flow patterns - show that the scaling laws are extremely complicated and difficult and experimental results can only be extrapolated to the original conditions by the help of a very extensive theoretical analysis. The attempt to work out such an analysis very soon shows that it has to comprise the analytical discription of the whole emergency core cooling process, i.e. the blowdown and the refilling phase and so it represents computer codes as they are given for example by the BRUCH-, the RELAP-, the SATAN- or the FLASH-code. That means a comprehensive and accurate extrapolation of the results from the experiment to reactor conditions has to be done with the help of a computer program which is able to predict as well the rundown of the test as the accident behaviour in the reactor. But such a program in addition needs several thermo-hydraulic data like heat transfer coefficients, DNB-delay time, critical mass flow rate and pressure drop behaviour of coolant pumps for a reliable analysis of the problem.

Due to these reasons the planning of the emergency core cooling program is born in our country by the conviction, that it is the most suitable way for quickly achieving the aim, to measure these parameters and data in different research activities and to evaluate at the same time a new computer program or to improve the known ones, which discribe the blowdown and refilling process well. These deliberations led to a research program in our country which is shown in fig. 4. The central position in this program - combining the different research activites - is taken by developing the computer codes. The experimental project can be divided in two large groups, one of them is concerned with the heat transfer and the fluid dynamic behaviour during the emergency cooling and the other one deals with the behaviour of the fuel rods. This second group is extensively treated by the National Research Center in Karlsruhe.

The heat transfer measurements, for clear physical boundary conditions sake, were subdivided in tests during the blowdown phase, and in tests during the refilling phase. In the blowdown phase the tests are carried out at the KWU Großwelzheim. They first were started with four-rod-clusters and now they are continued with 25-rod-clusters.

An example of the results gained at the four-rod-cluster is shown in fig. 5. There the heat transfer coefficients are plotted versus the time for two different heat fluxes. Before blowdown the heat flux was kept at 100% of the reactor power and in one case shown there the heat flux was immediately lowered to 40% with the onset of blowdown and in the other case there was a delay time of 1,2 sec for running down the power. In the test the cross section and the position of the break was varied. The figure shows a good agreement in the heat transfer coefficient for both heat flux behaviours, in spite of the fact that the wall temperatures are very different.

The evaluation of the tests shows that in the post-DNB-region the heat transfer coefficients amount to values between 100 and 500 $W/m^2K$ as shown in fig. 6, depending from the position and the cross section of the break. In this figure results are plotted with a break which corresponds to a steam pipeline rupture in a boiling water reactor. In the upper part of this figure the behaviour of the pressure, the temperature and the heat flux is shown. In the lower part the heat transfer coefficients for two axial positions of one rod in the bundle is plotted.

From fig. 7 one can see the mass flow rate and the steam quality at the outlet of the test section which correspond to the heat transfer coefficients in the figure before. The mass flow rate was measured by two different methods, firstly by a combination of a gamma-ray attenuation device and a flowmeter and secondly by an orifice. In this fig. 7 the calculated mass flow rate at the exit of the test section is also given as seen from the crosses. After about 10 seconds the measured mass flow rates lay below the calculated ones, which would mean that the code overpredicts the flow behaviour in the test section.

A very interesting feature furtheron was the observation that there is a rewetting of the rods due to a foaming of the fluid which would be seen from a drop in the rod temperature at temperatures well above the saturation value. In fig. 8 a histogram of all tests with which a rewetting occured is plotted versus the difference between rewetting temperature and saturation temperature. The upper values should not be taken too serious because the rewetting effect could not be analysed too exactly. Neglecting this upper values one can say that the temperature range at which rewetting occurs lays about 200 - 120 K above the saturation temperature.

More detailed information about the results of these measurements can be taken from different papers /2,3,4/ presented in national and international meetings. A comparison between this measured

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datas and calculated ones is given in /5/.

The refilling phase following the blowdown phase is researched in experiments which were carried out at the KWU in Erlangen. These measurements deal with the evaluation of the heat transfer coefficients and also give information about the refilling velocity and the rewetting of the fuel rods. For pressurized water reactors the test object was a rod cluster consisting of 340 rods, which was subdivided in 3 radial zones of different heat flux density. For measuring the emergency core cooling behaviour of a boiling water reactor at time two parallel rod bundles - each with 49 rods - are used as test objects. The results evaluated up to now give an extensive information about the whole parameter field to be expected during a loss-of-coolant-accident.

The complex geometry of the primary circuit and of the emergency cooling system was modelled in the test loop as shown in fig. 9. The core and the annular downcomer were imitated as a heated and unheated part of an U-tube. A very essential condition for the scaling is, that the physical connexion between the rising of the water level and the heat flux-respectively the steam production - is imitated as given in the real reactor. That means the heater rods must have the same dimensions and similar thermal behaviour as the fuel rods. The steam produced during flooding flows in the reactor through the primary loops. This resistance chain can be imitated in the test section by one single compensating resistance. The back pressure at the break, which corresponds to the containment pressure is imitated in the test with the help of a small pressure vessel connected with a steam generator. A detailed discription of the test arrangement can be taken from /6/.

In the test program the starting temperatures of the rods were varied between 500 and 850 $^{O}$C. The heat flux densities were between 4 and 8 W/cm$^2$, and the time course of the heat flux density was chosen according ANS standard. The back pressure, that is the containment pressure, was between 1 and 6 bar, and the liquid velocity in the core during refilling was varied between 3 and 18 cm/sec.

There is a strong influence of the containment pressure on the rewetting time in the core as shown in fig. 10. The reason for the sooner rewetting at higher containment pressures lays in the higher density of the steam and in the greater subcooling of the flooding water. At these higher containment pressures the vapor, produced in the core, has a smaller pressure drop in the tubes of the loop and so it can escape faster out of the upper plenum. The heat transfer coefficients - also shown in fig. 10 - lay around 100 and 300 $W/m^2K$. They show greater values with rising containment pressure.

As given in the KWU/PWR-reactors the test loop had a cold-leg and a hot-leg-injection. Tests were also done under conditions when the steam outlet from the upper plenum was totally blocked , and it was then seen that there is a balance between produced and - due to the hot-leg-injection - condensed steam. These blocked steam outlet conditions are an extrem unfavourable assumption for the behaviour in the reactor, but just these tests proved the assumption that the refilling velocity in the core is governed by the balance between removed or condensated and produced vapor. This is a simple, but very useful assumption for a theoretical treatment of the refilling process. Based on this experience a computer program called "WAK" /7/ was worked out by the KWU and a comparison between calculated and measured data for the refilling velocity gave very good agreement as shown in fig. 11. In this computer program it was assumed that the hot-leg-injection has a condensation efficiency of about 80%.

In fig. 12 a comparison is made between the measured heat transfer coefficient and data calculated from the Dittus-Boelter-equation, which is valid for pure steam convection and with data from the computer program LOCTA-R. As shown in this picture the theoretical calculated values according to Dittus Boelter or to the LOCTA-R program give conservative data in the post DNB region. This is due to the fact that the droplets in the vapor or respectively the water level which is already above the quenching border are neglected in these theoretical calculations.

More detailed information, especially about the influence of the heat flux, the refilling velocity and the temperatures at the beginning can be taken from /8,9/. Here only qualitatively should be mentioned that with growing refilling velocity the quenching-delay time becomes exponentially shorter and also the maximum temperatures go down extensively.

As already mentioned the refilling velocity of the water in the core can be strongly diminuated by the steam blocked in the upper plenum, an effect which is called "steam binding". Therefore continuing the above mentioned measurements in the KWU Erlangen a new test loop is under construction in which experiments concerning the steam binding effect will be conducted and in which especially it has to be doubtless proved that steam binding can be extensively avoided by the hot-leg-injection. The water injected in the hot leg, condensates - as mentioned - the steam in the upper plenum and so removes the reason for the blockage of the flow. First results about this condensing effect of the hot-leg-injection were gained in a big test carried out in the original pressure vessel of the nuclear power station in Borssele. These tests showed a condensation efficiency of 60 to 80%.

For a well rounded overall result it finally seems to be useful and desirable to imitate the blowdown phase in a complete pressurized water primary system. This mainly is interesting for a final testing of the worked out computer program. Due to this reason a research program was ordered at the Euratom Research Center in Ispra, which consists in a very sophisticated test loop and in conducting exactly specified blowdown measurements in this loop.

## 4. Fuel Rod Behaviour

As mentioned already in a previous chapter high intension is given to the research in fuel rod behaviour. First experimental results about the behaviour of zircon cladding under the conditions of a loss-of-coolant-accident were already presented by the KWU Erlangen. In a broad and extensive way, however, such tests are planned within the project nuclear safety of the National Research Center Karlsruhe. A group of about 50 collaborators is concerned with questions of the mechanical and thermal behaviour of fuel rods in and after the blowdown phase, of the reciprocal interaction between fuel rod failure and emergency cooling in the refilling and flooding phase and of the ductile and brittle material behaviour during fast temperature gradients. These experimental activities finally should lead to a computer program modelling this fuel rod behaviour under maximum coolant accident conditions. A detailed description of the fuel rod failure problems including a short layout of the plant research program is given in /10/.

With respect to the blowdown and refilling behaviour following a break it is important to know whether already in the blowdown phase flow disturbances due to plastic deformation of the cladding have to be expected or whether due to ballooning the heat flux to the coolant is influenced, which leads to a higher amount of stored heat in the fuel. The experiments planned in the refilling phase content as well out-of-pile as inpile tests but up to now these experiments are not yet planned in detail. An essential part of the out-of-pile experiments is the inquiry of the mutual action between cladding failure and emergency cooling behaviour. To research this, the National Laboratory in Karlsruhe is starting experiments with electrical heated rods having a zircaloy cladding with internal pressure and so simulating the fuel rod behaviour. In this connexion one has also to mention experiments which should clarify the influence of the ballooning on flow blockage and on the flooding behaviour. But also oxidation of the fuel cladding and oxygen embrittlement is researched at the National Laboratory in Karlsruhe.

In the post DNB period before rewetting - starting at a temperature of about 900°C - the onset of the zircon water reaction is to be expected. But already at slightly lower temperatures there may happen a ballooning or a ductile failure of the cladding which may influence the local temperature rise and may damage the fuel rods. At 1200°C the zircon water reaction is fully developed.

Due to the zircon water reaction an oxid layer is formed at the cladding which produces a concentration gradient for oxygen in the tube wall. At the same time hydrogen becomes free and so the conditions for a brittling of the cladding are given. Therefore one distinguishes between a ductile and a brittle cladding failure.

The most important parameters for the ductile cladding failure are

the inside pressure in the fuel rod,

the local and temporal temperature distribution during the loss-of-coolant-accident,

the equality of the cladding thickness and

the time and history of operation.

Up to now there are some data - especially from US-laboratories - giving the dependance of the temperature at which fuel rod failure occurs from the inside pressure in the fuel rod as shown in fig. 13. With growing pressure the temperature limit decreases at which the failure of the cladding occurs. In this figure mainly results gained from isothermal tests are plotted, i.e. in these tests at constant temperature the inside pressure was raised up to the failure of the cladding. One of the curves shows also results of transient tests. Furtheron there are pointed out the results of US manufactures and other US-laboratories. In the next picture - fig. 14 - the circumferential stretching of the ballooned and burst cladding is shown in dependance from the inside pressure. At a pressure of 28 until 42 bar there is a minimum of stretching which lays about at a value of 50% from the total circumference. At lower and higher pressure the stretching is going up to 100%. At pressures between 28 and 42 bars the failure temperatures lay between 815 and 980°C. In this temperature region zircaloy is present in the $\alpha$ - $\beta$ mixed cristal phase which differs from the pure $\alpha$ respectively $\beta$ phase by smaller fracture stretching.

Due to the ballooning of the rods a diminution of the flow cross section occurs, which produces an additional resistance for the coolant flow. Theoretically a circumferential stretching of about 30% amounts in a cooling channel blockage of 60% and at 70% stretching the channel would be completely blocked.

Up to now most tests in the literature were carried out with a constant heat input, i.e. with a linear temperature gradient in time. Already these tests showed that the temporal temperature gradient has a clear influence on the failure temperature and on the amount of stretching. Nonlinear time-temperature conditions were not yet tested.

In a brittle failure the cladding bursts almost without any stretching or it is destroyed due to the high thermal tensions during the quenching which is called shattering. A brittle failure is to be expected, if in the circaloy cladding an oxygen-rich phase occurs. Then due to steam oxidation in the outer surface of the cladding a $ZrO_2$ layer is formed below which an $O_2$ stabilized phase can exist. The most important influencing parameters on the change of the stretching properties are

- the time, the cladding is staying at constant temperature,
- the temperature-time course of a loss-of-coolant accident,
- the wall thickness of the cladding.

Generally one can say that with a thickness of the $ZrO_2$ layer of more than 18% brittle failure is to be expected.

## 5. Licensing Rules Concerning Emergency Core Cooling

The German reactor safety criteria for emergency core cooling are in some parts similar to the US interims acceptance criteria. The layout of the emergency core coolant system must be so that the core temperatures can be kept at low values for long time. Especially

1. the calculated maximal cladding temperatures must not exceed 1.200 $^{\circ}$C,
2. the calculated oxydized thickness of the cladding at any position must not reach the value of 17% of the real wall thickness,
3. not more than 1% of the whole zircon material in the cladding is allowed to undergo the zircon water reaction,
4. due to failure of the cladding not more than 10% of the iodine and not more than 0,1% of the solid fission products are allowed to become free,
5. no change to such an extent in the geometry of the reactor core may occur that a sufficient cooling of the core is prevented.

The critical mass flow rate is calculated according to the moody equation /11/ and immediately after reaching the critical heat flux stable film boiling has to be assumed. For the heat transfer coefficients during the blowdown and the refilling period approved measurements can be taken in account. If measurements are not available. The heat transfer coefficients have to be calculated with the Dougall-Rohsenow-equation /12/ or with the Groeneveld-equation /13/ depending from its validity region.

In the emergency core cooling analysis it is assumed that one cooling system fails, another one is just under repair, and the water flowing to the broken leg does not reach the core. So only one complete emergency cooling system is regarded to be available for refilling the core. Another one only through the unbroken leg feeds into the core, if there is a cold-leg and a hot-leg-injection. So totally spoken it is assumed that one and a half emergency cooling systems are available in the worst case, which have to guarantee that the temperatures in the core do not exceed unallowable boundaries.

The licensing procedure for the first big pressurized water reactor Biblis A showed the important role of the hot-leg-injection in emergency core cooling, which German pressurized water reactors are equipped with. Due to the condensing effect of the hot-leg-injected water the refilling process is expected to be guaranteed in such a way that the maximum temperatures lay well below 1000 $^{\circ}$C, which amounts in a negligible number of failed fuel rods. So also one has not to expect a worth mentioning zircon water reaction.

For the layout of all safety devices against loss-of-coolant-accident it has to be assumed that the break can occur at any operation condition which is in the specification for running the reactor, i.e. also certain transients have to be included. The blowdown calculation is done with the German computer programs BRUCH-D and RELAP 4. For evaluating the forces acting on the core structure during the very first seconds after the break the core structure itself is assumed be behave stiff. This certainly is a conservative assumption with respect to the real conditions.

As known after refueling collapsed claddings were found 1972 in the nuclear reactor Beznau and in the Westinghouse pressurized water reactors Ginna and Robinson II. Today one knows that the reason for this collapsed cladding was a densification due to irradiation of the $UO_2$ fuel. In the meantime this densification showed to be independend from the rod power and from the fuel temperature and occurs after about 2000 hours of operation. This densification produces a larger gap between the cladding and the pellets, which reduces the heat transfer from the fuel to the cladding and enlarges the stored heat in the fuel during LOCA. The pellets also become shorter in axial direction which produces empty spaces and local flux peaks.

In German nuclear power stations up to now no collapsed cladding or fuel densification was observed. This is true as well for pressurized as for boiling water reactors. It is assumed that the reason for this fact is a sintering process which produces pellets of higher density and great thermal stability. Therefore in the German licensing procedure the fuel densification up to now has not to be taken in account. Nevertheless in our country also research activities for understanding the fuel densification and for the optimalisation of the sintering process were encouraged. The aim of the planned activities is to study the porosity of the sintered fuel under certain boundary conditions and to test the change intensity during irradiation. So by optimalisation of the technological parameters a fuel should be produced which gives no rise to densification.

## 6. Final Remarks

Last not least there has to be mentioned one problem which does not seem to belong to the reactor safety research, i.e. the problem of making available all the achieved scientific and technical new knowledge. There one has not only to think about how to inform the technical people like manufacturer, consultants, supervisors and regulatory organisations, but there is mainly the problem how to inform in a general understanding manner the public. For informing the technical people there are deliberations at the National Research Center Karlsruhe to work out an information system for reactor safety know-how, which is called RESI.

This system should help to work up all informations which are essential for the licensing procedure and should make the results and the information easy accessible to all in nuclear power plants . involved institutions.

The information activities for the public not only have to mediate general understandable knowledge and so to liberate the layman from the myth or from the fear of the atomic energy it also has to show how the risks coming from the nuclear power stations are related to the other nat~~ional~~ural and technical risks which we are exposed to in our daily life.

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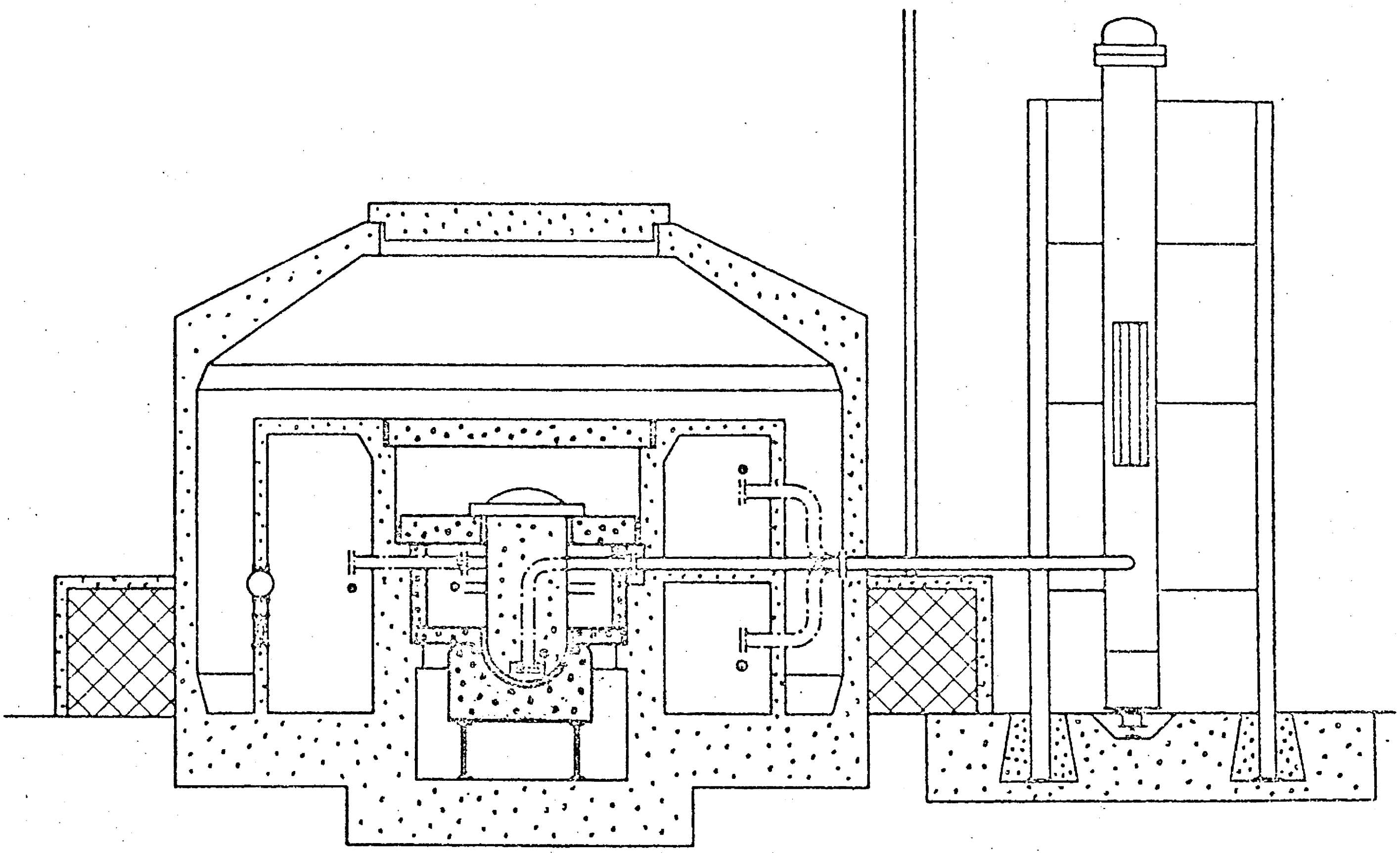


FIG. 1 : TEST CONTAINMENT ( BATELLE )

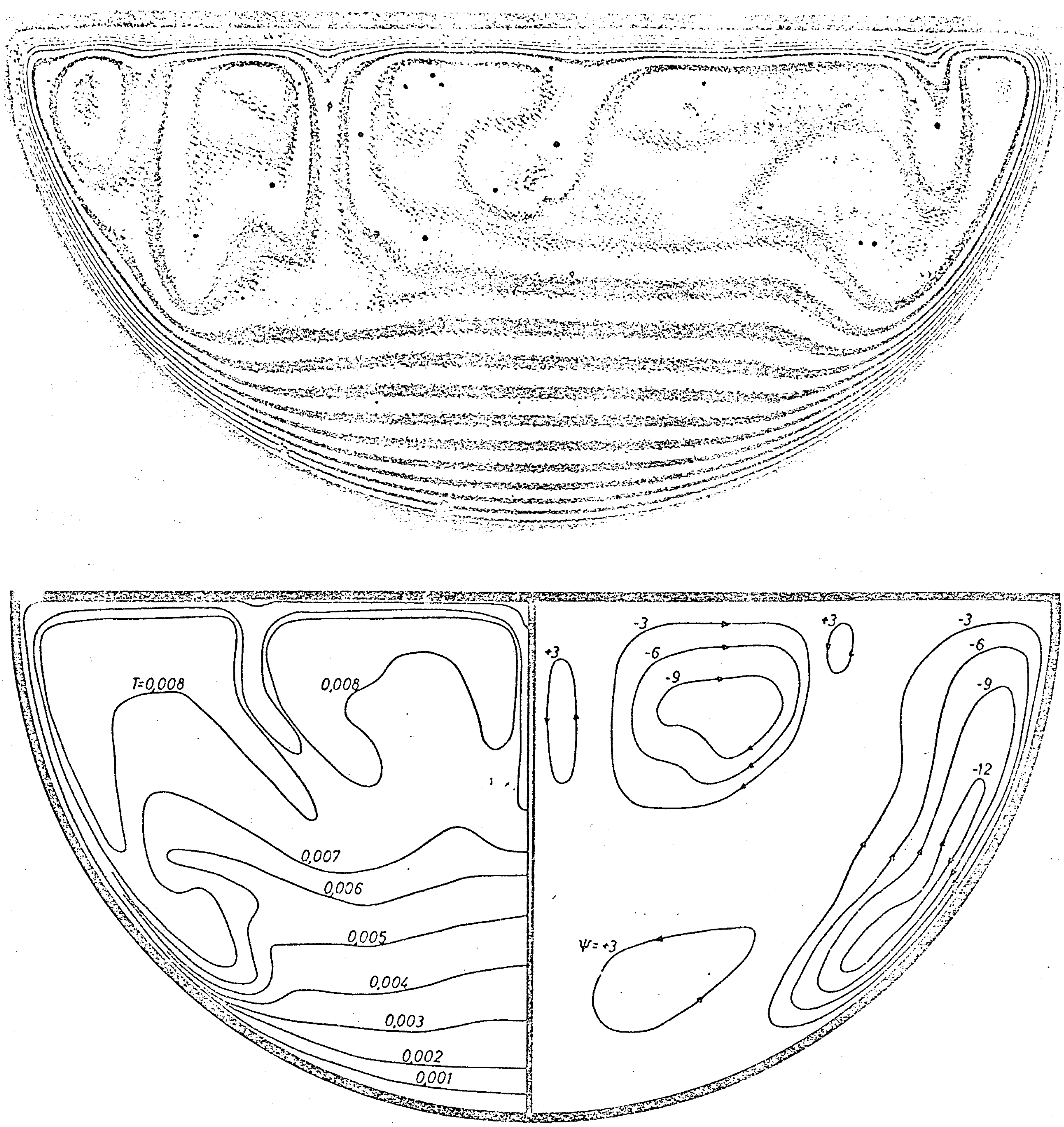



FIG. 2 : THERMODYNAMIC BEHAVIOUR OF A MOLTEN CORE
ISOTHERMS IN THE MELT

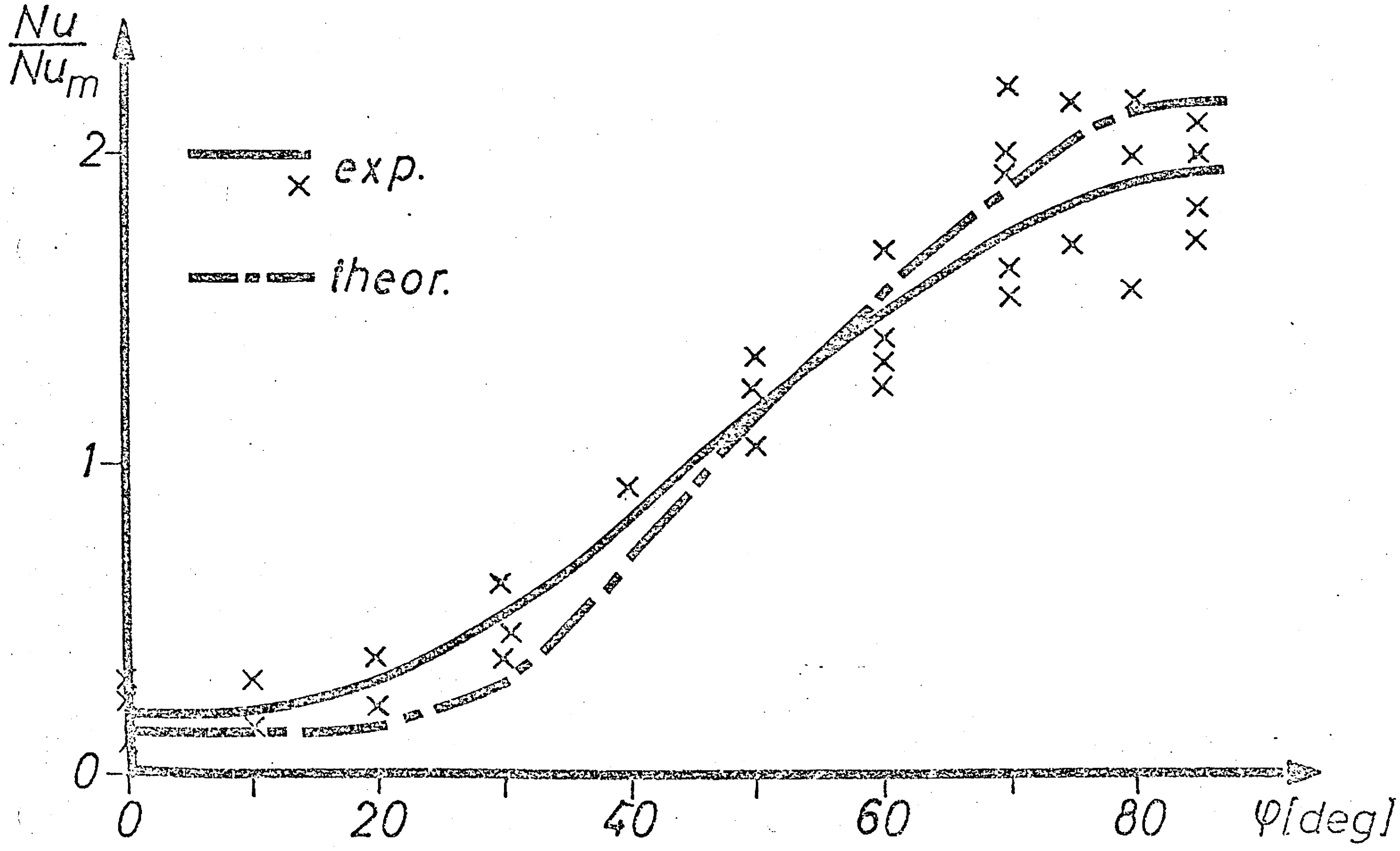



FIG. 3 : HEAT TRANSFER FROM CORE - MELTING TO
REACTOR - PRESSURE VESSEL

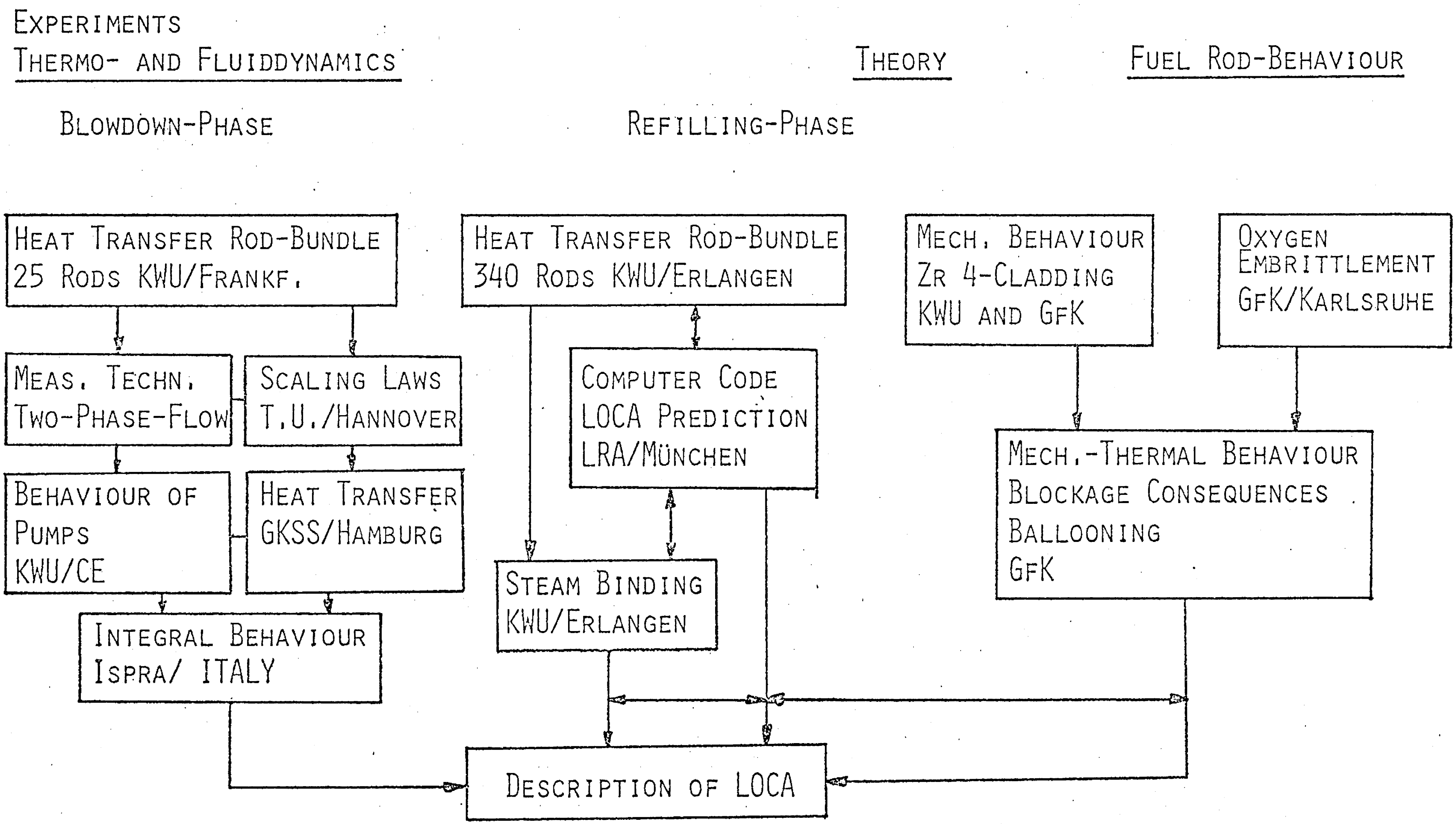



Fig. 4 : German Research Program : Emergency Core Cooling and Fuel Rod Behaviour

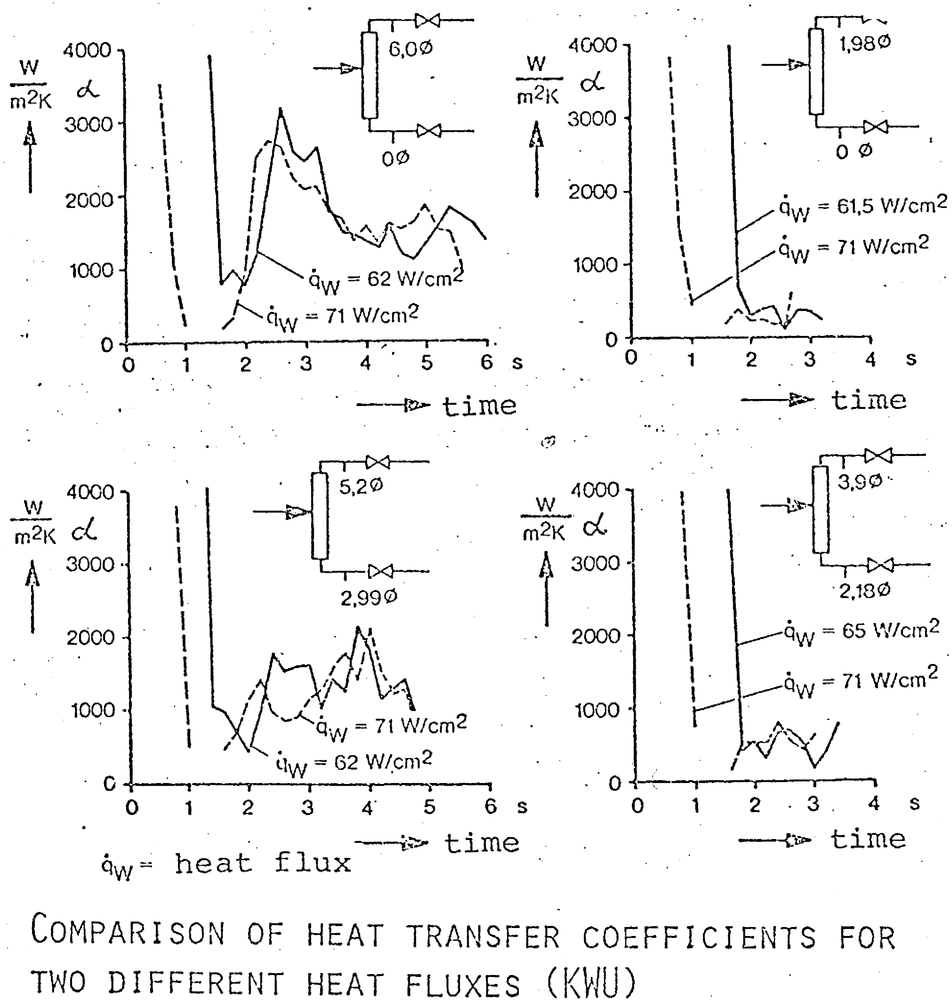
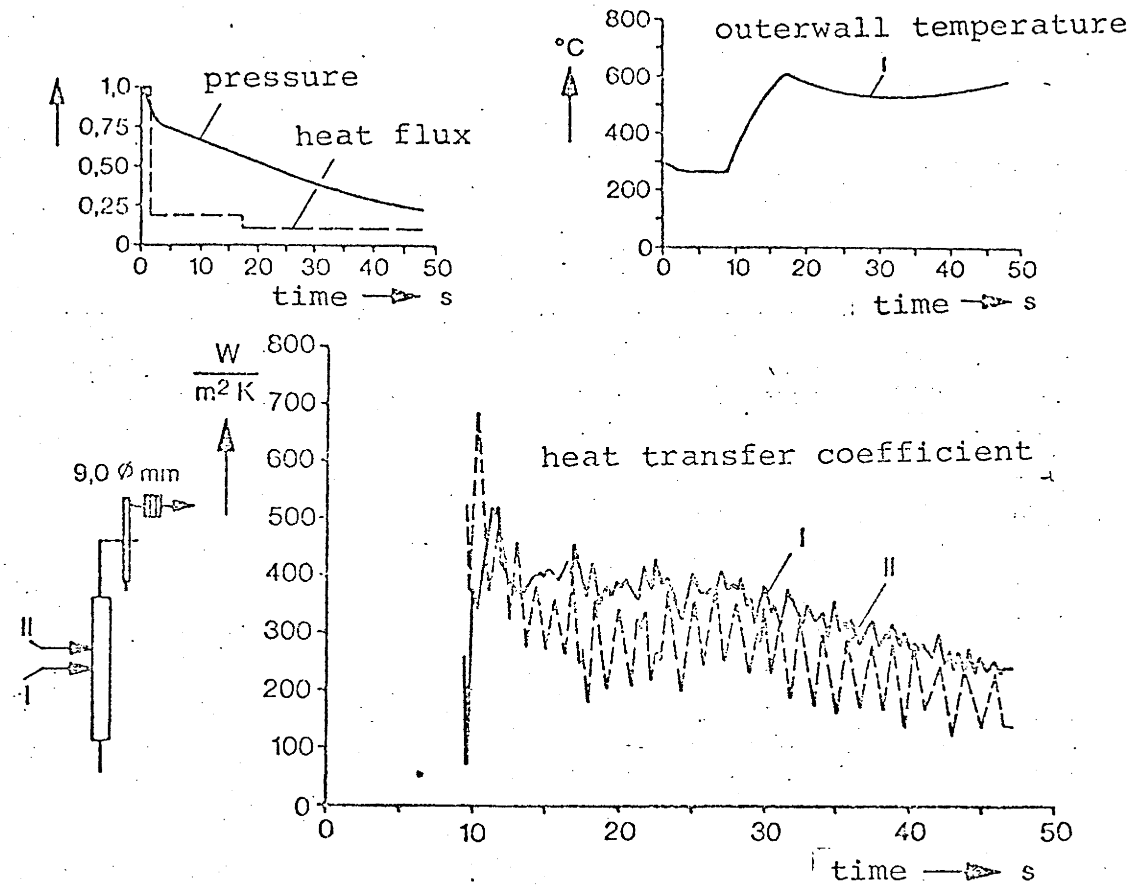



FIG. 5 COMPARISON OF HEAT TRANSFER COEFFICIENTS FOR TWO DIFFERENT HEAT FLUXES (KWU)




Experimental results, 4-rod bundle (KWU)

F 73 102

Fig. 6

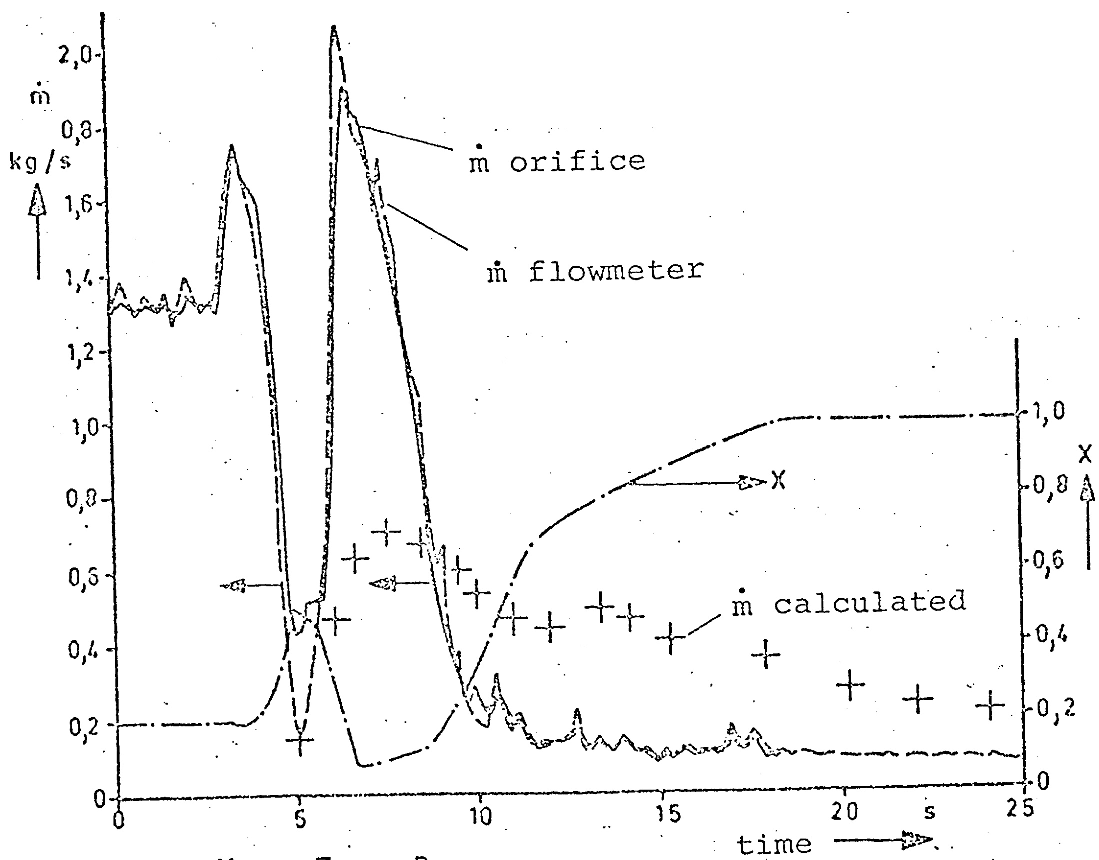
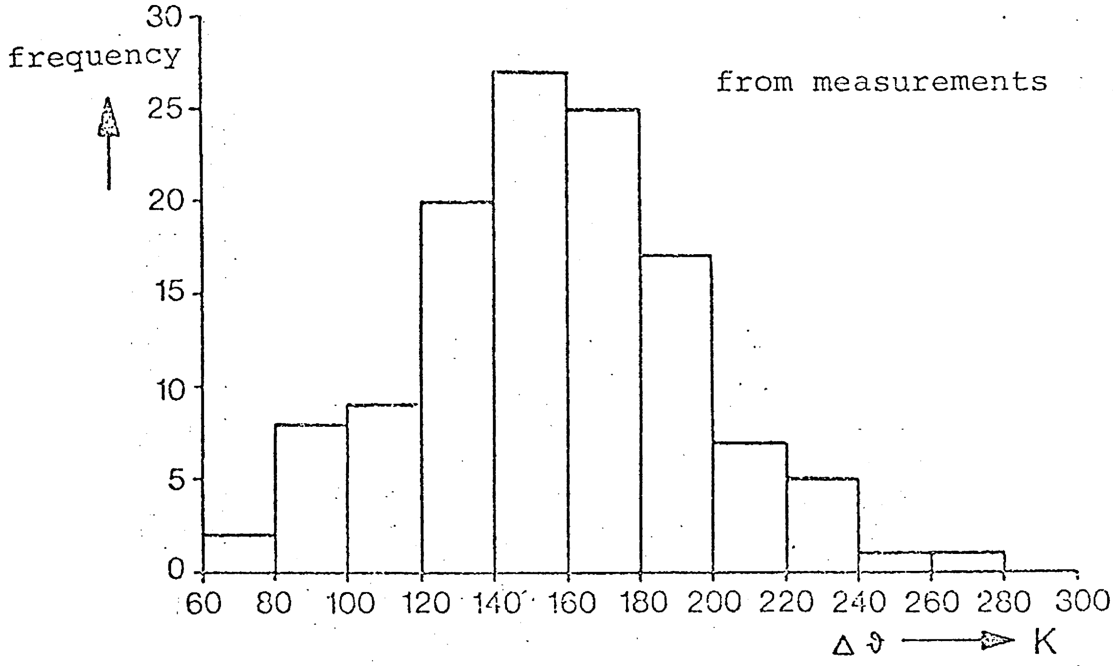
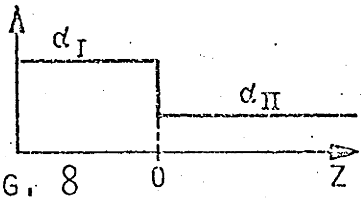



Fig. 7 Mass Flow Rate and
Void Fraction at Test Section Exit (KWU)




$$\frac{1-\sqrt{1+A^2}}{1+\sqrt{1+\frac{\alpha_{II}}{\alpha_I}A^2}} = \frac{\vartheta_{+\infty}-\vartheta_0}{\vartheta_{-\infty}-\vartheta_0}\,;\quad A^2 = \frac{4\alpha_I\,\lambda}{\delta\,\rho^2\,c^2}\,\frac{1}{u^2}$$

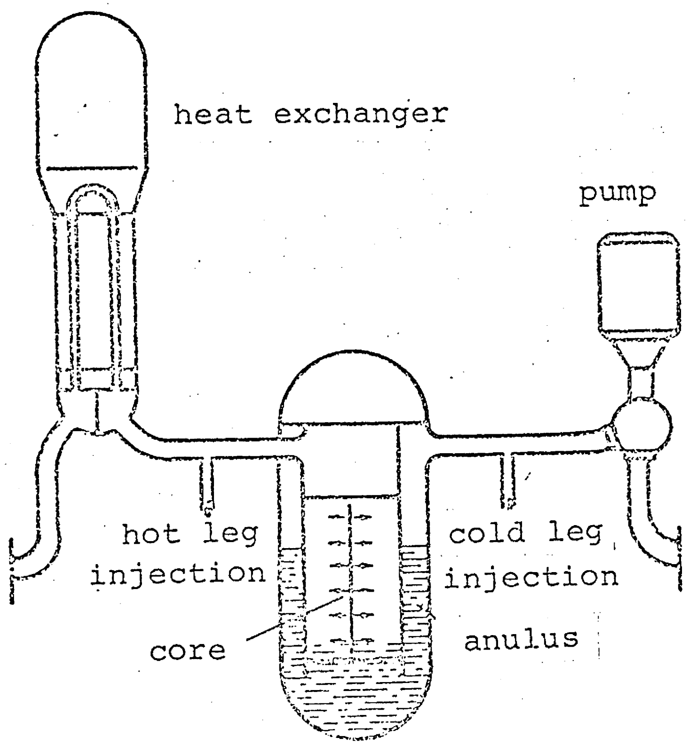
FIG. 8

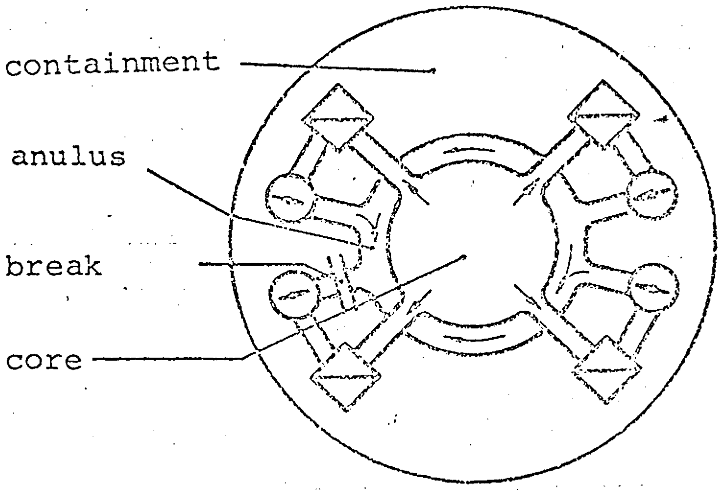
REWETTING TEMPERATURES $\Delta\vartheta = \vartheta_W - \vartheta_S$

AT FOAMING FLUID (KWU)

F 73 107

HEIGHT SCHEME OF PWR




principle design
of PWR-loop

LAYOUT OF LOOP

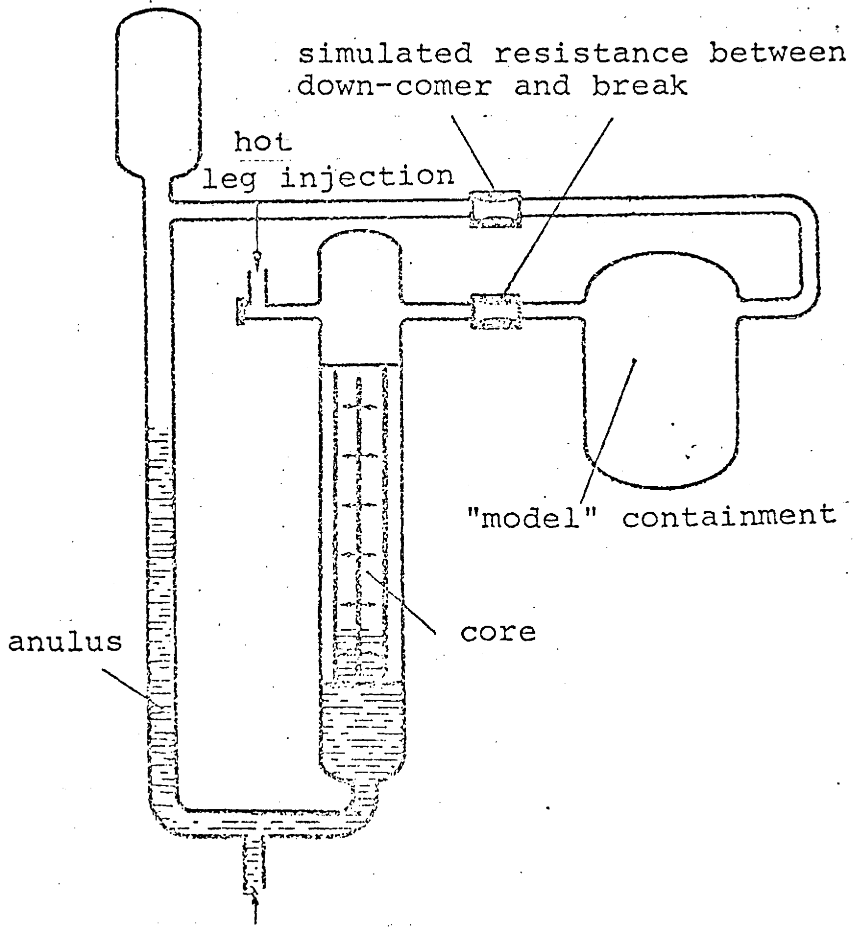



FIG. 9 MODEL-LOOP OF PWR (KWU)

FLOODING (ROD-BUNDLE) RUN-NO. 83/95

| | | | |
|---|---|---|---|
| mass flow rate: | 20,9 $m^3/h$ | | |
| theoretical velocity: | 4,5 cm/s (cold) | pressure: | 1,0/4,5 bar |
| theoretical velocity: | 4,5 cm/s (hot) | rod-power: | 6,0 kW |
| temperature level: | 650,0 °C | outlet resistance: | 15 |

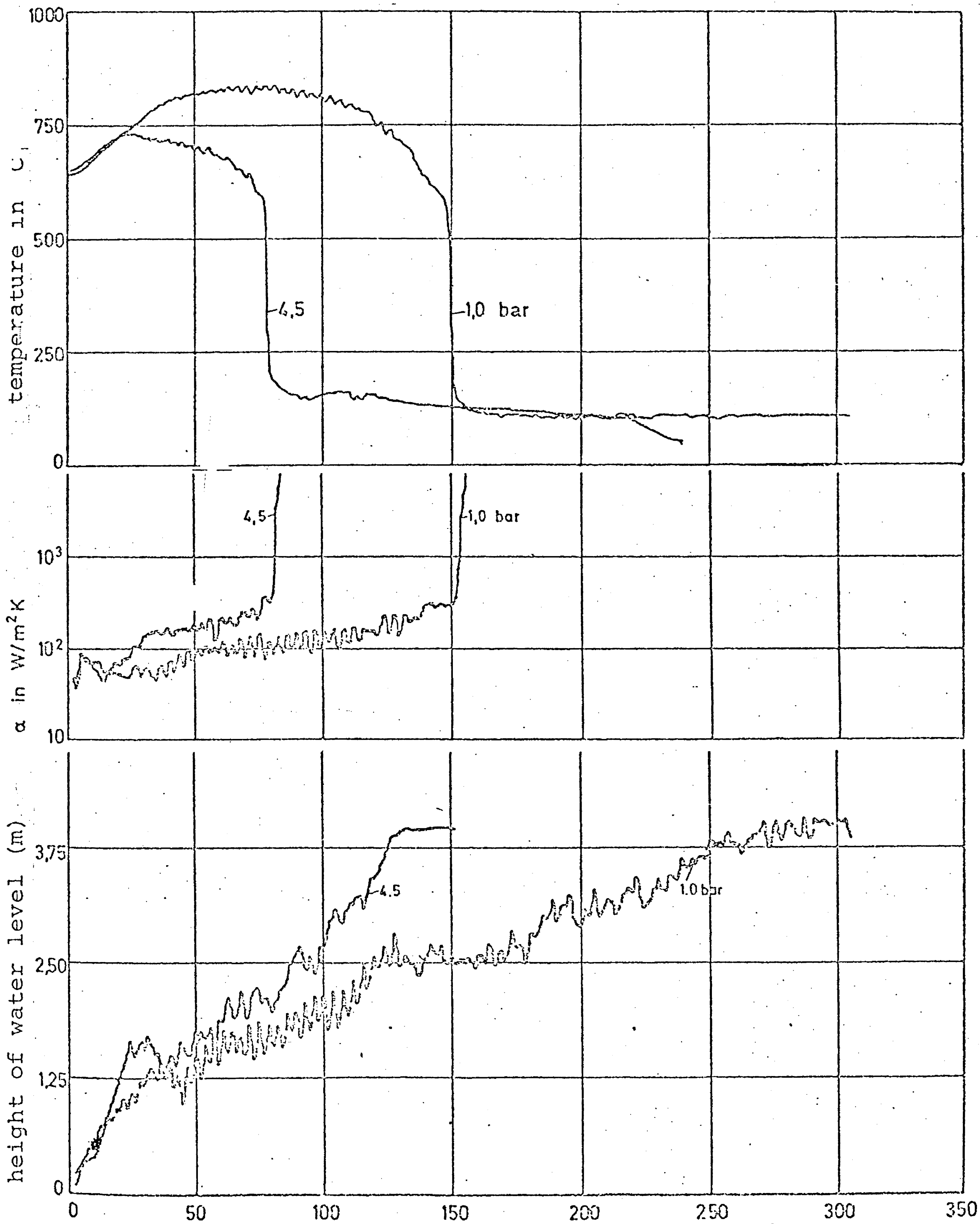



FIG. 10 INFLUENCE OF PRESSURE ON FLOODING (KWU)

FLOODING (ROD-BUNDLE)

| | | |
|---|---|---|
| WK (CM/S) | = | 9.0 |
| Q2SMAX (W/CM2) | = | 8.2 |
| THUELL,0 (GRD C) | = | 650.0 |
| TKM,0 (GRD C) | = | 30.0 |
| P0 (BAR) | = | 4.5 |
| ZETA | = | 15.0 |

TOP AND BOTTOM FLOODING

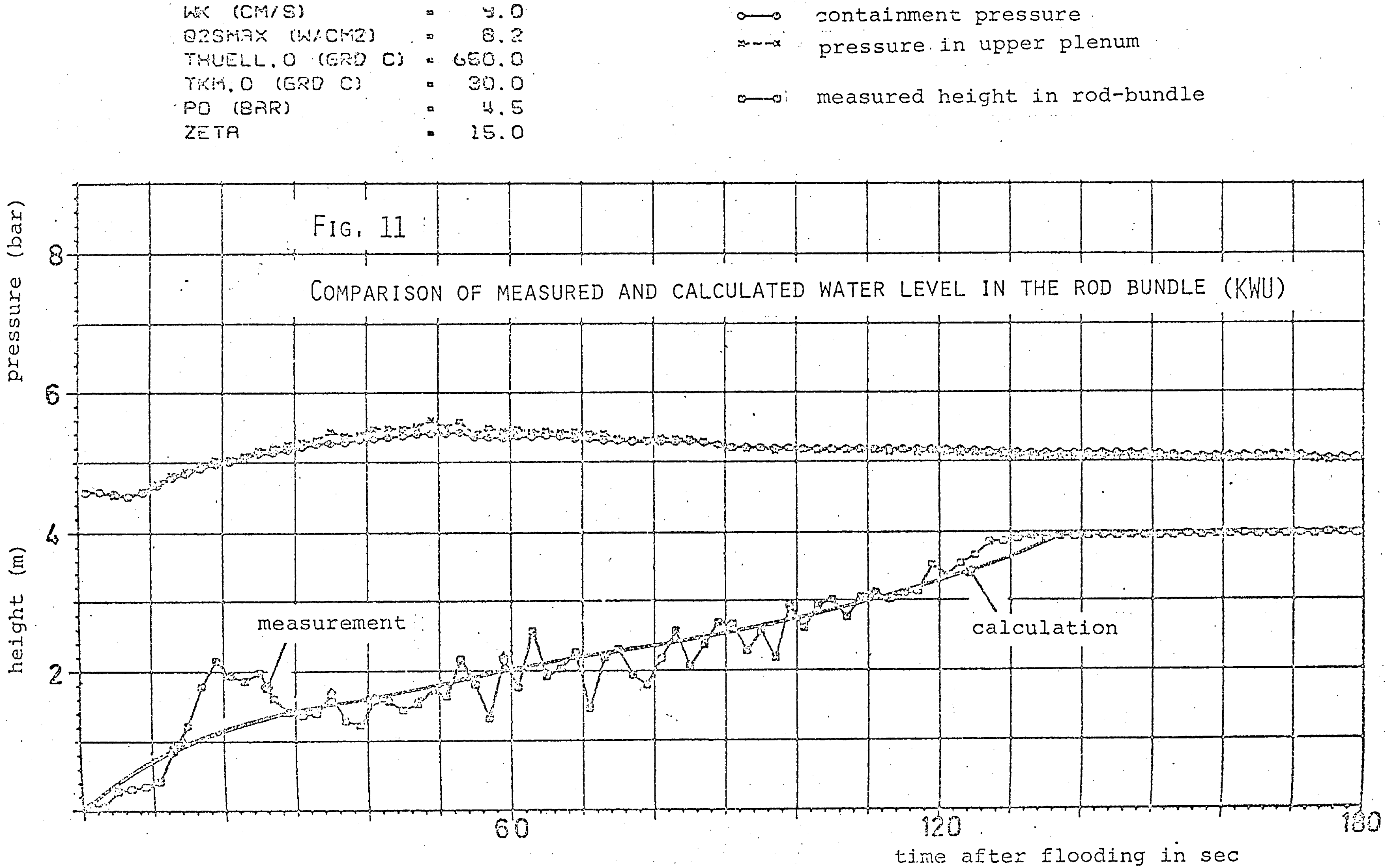



FIG. 11

COMPARISON OF MEASURED AND CALCULATED WATER LEVEL IN THE ROD BUNDLE (KWU)

FLOODING ROD-BUNDLE

RUN-NO. 83

mass flow rate: 20,9 $m^3/h$
theoretical velocity: 4,5 cm/s (cold)
theoretical velocity: 4,5 cm/s (hot)
temperature level: 650,0 °C

pressure: 4,5 bar
rod-power: 6,0 kW
outlet resistance: 15

rod Nr.:6/E4

date 25.5.73

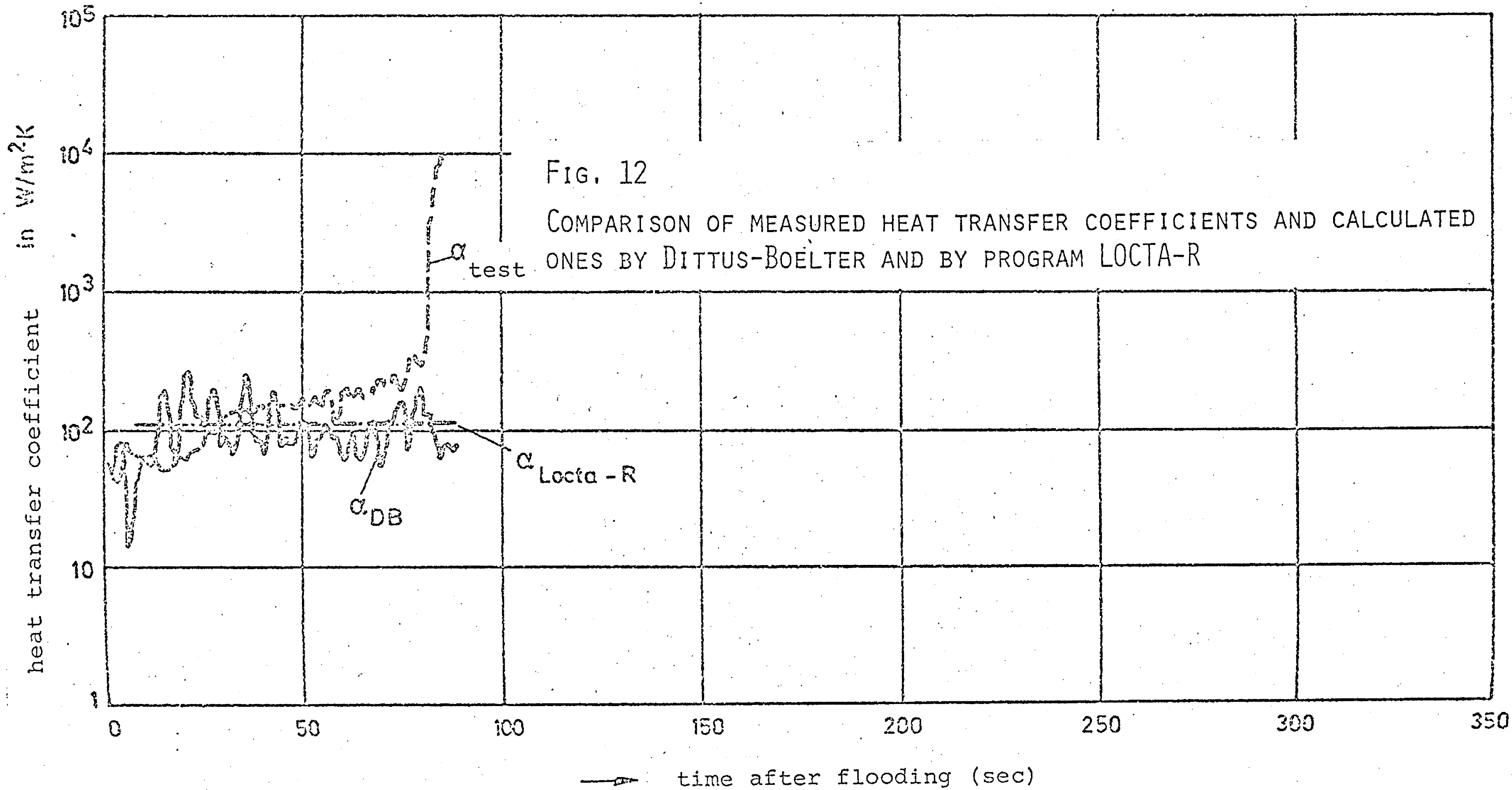



FIG. 12

COMPARISON OF MEASURED HEAT TRANSFER COEFFICIENTS AND CALCULATED ONES BY DITTUS-BOELTER AND BY PROGRAM LOCTA-R

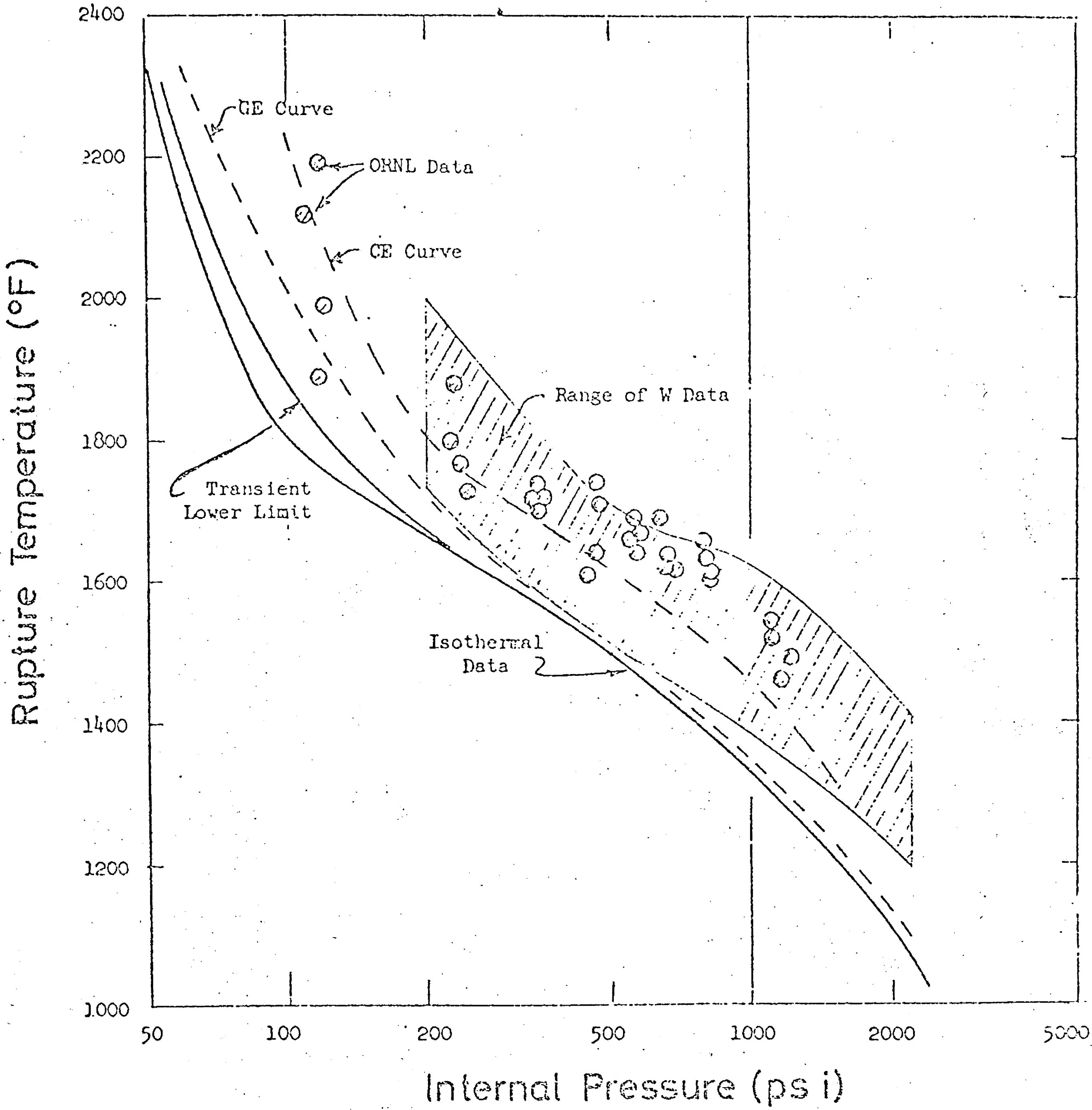



FIG. 13 : CIRCUMFERENTIAL STRETCHING OF CLADDING

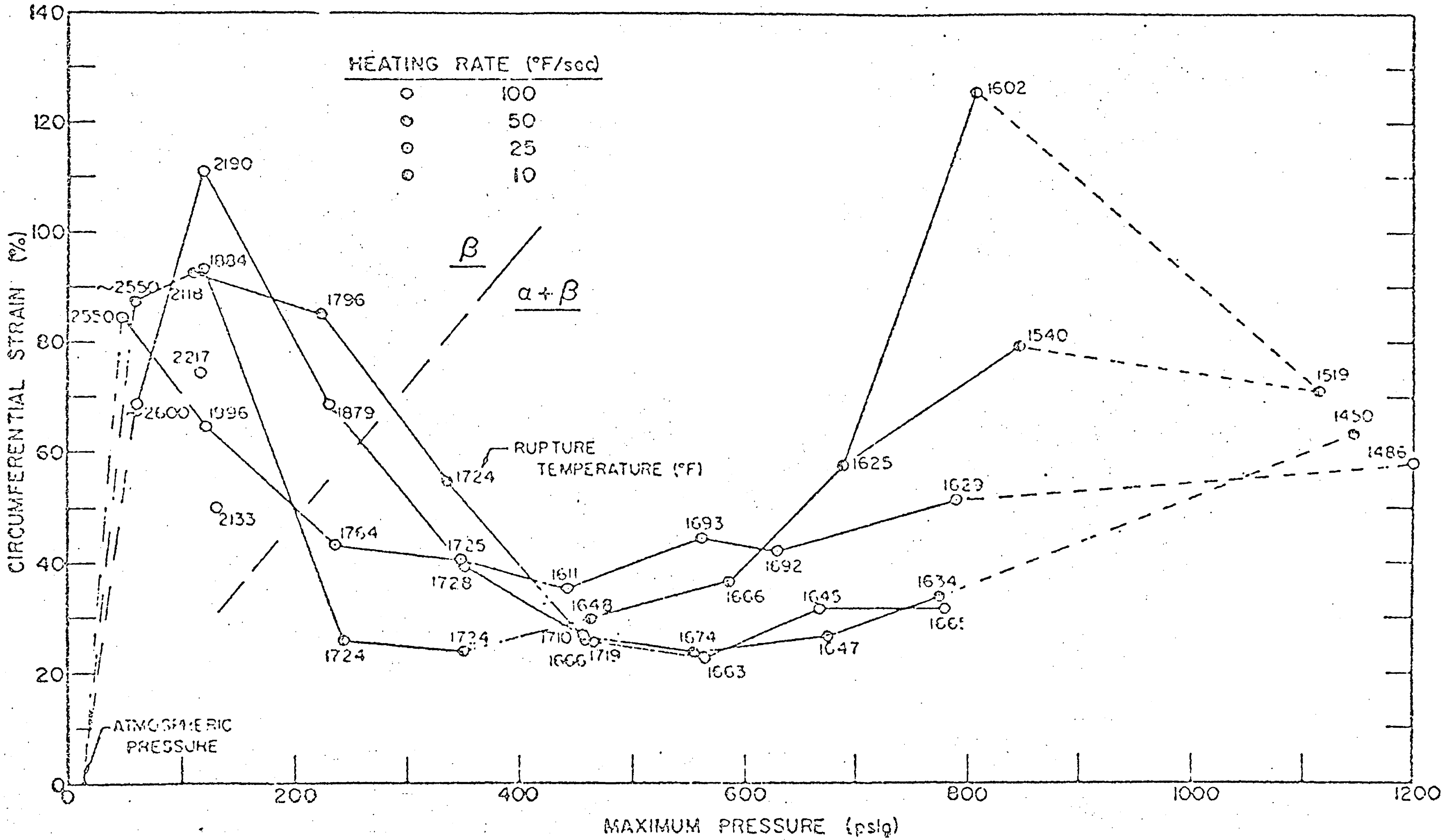



FIG. 14: FUEL ROD FAILURE AS FUNCTION OF ROD - INSIDE PRESSURE

TABLE 1

## RESEARCH ACTIVITIES TO AVOID ACCIDENTS

### 1. SYSTEM ANALYSIS

REACTOR - DYNAMICS AND THERMO - HYDRAULICS
RELIABILITY OF COMPLEX SYSTEMS

### 2. DESIGN AND STRENGTH

DESIGN OF REACTOR PRESSURE VESSELS
STRENGTH EXPERIMENTS AT PRIMARY SYSTEM COMPONENTS
QUALITY - GUARANTEE BY TESTING METHODS

### 3. PLANT SUPERVISION

CRACK AND FAILURE BEHAVIOUR
REPEATED TESTS

### 4. PERSONNEL TRAINING

TRAINING OF OPERATORS
ANALYSIS OF HUMAN FUNCTIONS

### 5. HYPOTHETICAL ACCIDENTS

BURST SECURITY FOR PRESSURE VESSEL
BURST SECURITY FOR COMPONENTS

TABLE 2

RESEARCH ACTIVITIES TO MASTER THE EFFECTS OF ACCIDENT

## 1. Blowdown

STRAIN IN REACTOR COMPONENTS
PRESSURE - SUBPRESSION
FLOW BEHAVIOUR IN CONTAINMENT

## 2. Emergency Core Cooling

HEAT TRANSFER IN THE CORE
LOOP BEHAVIOUR
FUEL ROD BEHAVIOUR

## 3. Hypothetical Accident Coremelting

THERMODYNAMIC AND HEAT TRANSFER PROBLEMS
METALLURGICAL AND CHEMICAL QUESTIONS
PHYSICAL PROPERTIES AND RADIOACTIVE RELEASE

## 4. Actions from Outside

EARTHQUAKE
AIRPLANE CRASH
GASEOUS EXPLOSIONS

TABLE 3

## RESEARCH ACTIVITIES TO REMOVE CONSEQUENCES OF AN ACCIDENT

### 1. PROTECTION AGAINST EXTREM ACCIDENTS

MANIPULATORS FOR EMERGENCY CASE
DECONTAMINATION
REMOVING OF RADIOACTIVE WASTE

### 2. PROTECTION AGAINST IRRADIATION

RELEASE DURING NORMAL OPERATION AND DURING REPAIR
PROPAGATION OF RADIOACTIVITY
LONG - TIME ENVIRONMENTAL RADIOACTIVE LOADING