EUR 352.e

EUROPEAN ATOMIC ENERGY COMMUNITY – EURATOM
UNITED STATES ATOMIC ENERGY COMMISSION – USAEC

TWO-PHASE FLOW PROBLEMS

1963

Proceedings of the Meeting
of the Working Group Heat Transfer
sponsored by
the Euratom-United States Joint R. & D. Board
Brussels, October 29, 30, 31, 1962

LEGAL NOTICE

This report can be obtained, at the price of Belgian Francs 175.—, from: PRESSES ACADEMIQUES EUROPEENNES - 98, Chaussée de Charleroi, Brussels 6.

Please remit payments:

— to BANQUE DE LA SOCIETE GENERALE (Agence Ma Campagne) - Brussels - account No 964.558,

— to BELGIAN AMERICAN BANK AND TRUST COMPANY - New York - account No 121.86,

— to LLOYDS BANK (Foreign) Ltd. - 10 Moorgate - London, E.C.2,

giving the reference: "EUR 352.e - Two-Phase Flow Problems".

Printed by Snoeck-Ducaju & Son,
Brussels, June 1963.

EUR 352.e

TWO-PHASE FLOW PROBLEMS

European Atomic Energy Community - EURATOM
United States Atomic Energy Commission - USAEC
Proceedings of the Meeting of the Working Group Heat Transfer sponsored by the Euratom-United States Joint R. & D. Board
Brussels, October 29, 30, 31, 1962
Brussels, June 1963 - pages 152

The USAEC-EURATOM Joint Board arranged a meeting in Brussels on 29, 30 and 31 October 1962 for the discussion of problems bound up with double phase. The meeting was attended by both USAEC contractors and American and European contractors of the Joint Board.

Working meetings of this kind have already been held on such matters as the dynamics of boiling-water reactors, fuel and plutonium recycling. This meeting was fully justified by the considerable attention which has been paid to the

EUR 352.e

TWO-PHASE FLOW PROBLEMS

European Atomic Energy Community - EURATOM
United States Atomic Energy Commission - USAEC
Proceedings of the Meeting of the Working Group Heat Transfer sponsored by the Euratom-United States Joint R. & D. Board
Brussels, October 29, 30, 31, 1962
Brussels, June 1963 - pages 152

The USAEC-EURATOM Joint Board arranged a meeting in Brussels on 29, 30 and 31 October 1962 for the discussion of problems bound up with double phase. The meeting was attended by both USAEC contractors and American and European contractors of the Joint Board.

Working meetings of this kind have already been held on such matters as the dynamics of boiling-water reactors, fuel and plutonium recycling. This meeting was fully justified by the considerable attention which has been paid to the

EUR 352.e

TWO-PHASE FLOW PROBLEMS

European Atomic Energy Community - EURATOM
United States Atomic Energy Commission - USAEC
Proceedings of the Meeting of the Working Group Heat Transfer sponsored by the Euratom-United States Joint R. & D. Board
Brussels, October 29, 30, 31, 1962
Brussels, June 1963 - pages 152

The USAEC-EURATOM Joint Board arranged a meeting in Brussels on 29, 30 and 31 October 1962 for the discussion of problems bound up with double phase. The meeting was attended by both USAEC contractors and American and European contractors of the Joint Board.

Working meetings of this kind have already been held on such matters as the dynamics of boiling-water reactors, fuel and plutonium recycling. This meeting was fully justified by the considerable attention which has been paid to the

question of double phase flow under the auspices of the Joint Board, which was anxious to take stock of the progress already achieved under the current contracts and to pinpoint the direction in which research trends were moving before the launching of the second five-year plan.

Furthermore, one of the chief benefits to be derived from the Joint Program is that it makes it possible, on the technical level, to set up a scheme of intimate collaboration between the various research workers and laboratories engaged in work in the same field. The results stemming from the double-phase program, in both quantity and quality, provided a guarantee that a thorough discussion of the problems at issue would prove most useful.

This report contains a summary of the contributions made by those present at the meeting.

question of double phase flow under the auspices of the Joint Board, which was anxious to take stock of the progress already achieved under the current contracts and to pinpoint the direction in which research trends were moving before the launching of the second five-year plan.

Furthermore, one of the chief benefits to be derived from the Joint Program is that it makes it possible, on the technical level, to set up a scheme of intimate collaboration between the various research workers and laboratories engaged in work in the same field. The results stemming from the double-phase program, in both quantity and quality, provided a guarantee that a thorough discussion of the problems at issue would prove most useful.

This report contains a summary of the contributions made by those present at the meeting.

question of double phase flow under the auspices of the Joint Board, which was anxious to take stock of the progress already achieved under the current contracts and to pinpoint the direction in which research trends were moving before the launching of the second five-year plan.

Furthermore, one of the chief benefits to be derived from the Joint Program is that it makes it possible, on the technical level, to set up a scheme of intimate collaboration between the various research workers and laboratories engaged in work in the same field. The results stemming from the double-phase program, in both quantity and quality, provided a guarantee that a thorough discussion of the problems at issue would prove most useful.

This report contains a summary of the contributions made by those present at the meeting.

EUR 352.e

EUROPEAN ATOMIC ENERGY COMMUNITY - EURATOM
UNITED STATES ATOMIC ENERGY COMMISSION - USAEC

TWO-PHASE FLOW PROBLEMS

1963

Proceedings of the Meeting
of the Working Group Heat Transfer
sponsored by
the Euratom-United States Joint R. & D. Board
Brussels, October 29, 30, 31, 1962

CONTENTS

LIST OF PARTICIPANTS

U.S. PARTICIPANTS

Messrs. SCROGGINS (USAEC, Washington)
FERREL (North Carolina State College)
HOFFMAN (O.R.N.L.)
LEVY (General Electric Company)
MARCHATERRE (A.N.L.)

Messrs. MOISSIS (M.I.T.)
WALLIS (Dartmouth College)
HELFRICH (USAEC, Brussels)
LABOWITZ (USAEC, Brussels)

EUROPEAN PARTICIPANTS

Firm: AEG, *Germany*
Mr. MATTERN

Firm: ALSTHOM, *France*
Messrs. DOMENJOUD
ROBERT

Firm: G.E.N.G. *France*
Messrs. FABRÉGA
MONDIN
SAUNIER
VILLENEUVE

Firm: CISE *Italy*
Mr. SILVESTRI

Firm: FIAT *Italy*
Mr. PREVITI

Firm: M.A.N. *Germany*
Messrs. AGENA
BRADFUTE
MAYINGER

Firm: SNECMA *France*
Messrs. FOURÉ
MOUSSEZ

Firm: SORIN *Italy*
Mr. CAMPANILE

Firm: T.H.E. *The Netherlands*
Messrs. BOGAARDT
SIMON-THOMAS
SPIGT

Firm: THOMSON-HOUSTON *France*
Messrs. BEURTHERET
DEMANGE
DOUGUET
LE FRANC

UNIVERSITE DE GRENOBLE *France*

Mr. CRAYA

EURATOM

Messrs. BACKS
BAHBOUT
BERTOLETTI
EHRENTREICH
GRASS
KAUT
KRUYS
LÜTTMANN
MORIN
MÖRK-MÖRKENSTEIN
SIEBKER

INTRODUCTION

The USAEC-EURATOM Joint Board arranged a meeting in Brussels on 29, 30 and 31 October 1962 for the discussion of problems bound up with double phase. The meeting was attended by both USAEC contractors and American and European contractors of the Joint Board.

Working meetings of this kind have already been held on such matters as the dynamics of boiling-water reactors, fuel and plutonium recycling. This meeting was fully justified by the considerable attention which has been paid to the question of double phase flow under the auspices of the Joint Board, which was anxious to take stock of the progress already achieved under the current contracts and to pinpoint the direction in which research trends were moving before the launching of the second five-year plan.

Furthermore, one of the chief benefits to be derived from the Joint Program is that it makes it possible, on the technical level, to set up a scheme of intimate collaboration between the various research workers and laboratories engaged in work in the same field. The results stemming from the double-phase program, in both quantity and quality, provided a guarantee that a thorough discussion of the problems at issue would prove most useful.

The following report contains a summary of the contributions made by those present at the meeting.

I. USAEC TWO-PHASE FLOW PROGRAM

R.M. SCROGGINS
*Engineering Development Branch, Office of Nuclear Technology,
Division of Reactor Development, USAEC, Washington 25, D.C., USA*

This summary will present a general description of the USAEC Two-Phase Flow Program; its purpose, present program, and future objectives. This program is under the direction of the Commission's Office of Nuclear Technology and includes both direct AEC contracts and also those U.S. projects in the field under the Joint U.S.-Euratom Research and Development Program. A brief description of the Commission's overall Heat Transfer Program is also included.

The primary objective of the Commission's Nuclear Technology Heat Transfer Program is to develop fundamental and applied information which will permit the building of civilian power, advanced space and military reactor systems having higher power density, improved thermal efficiencies and power conversion, greater reliability and safety, and ultimate lower costs.

Most of the initial work supported by the AEC in the area of two-phase flow and boiling heat transfer was highly projectized and was a part of the Commission's boiling water reactor program. The majority of the results of these studies were based on system geometries and conditions and were usually correlated by empirical techniques. The results of this work were both interesting and quite useful to the beginning of our understanding of two-phase flow; but as the complexity of each individual project grew it became more difficult to coordinate the individual heat transfer studies.

In order to meet the objectives of the over-all heat transfer program with respect to present and proposed systems, it was decided that a single coordinated program of research on two-phase flow was necessary. It is obvious that results of such a program will be useful and necessary to the optimization of boiling water reactor systems; however, these studies should be most useful and probably necessary for the development of reactor systems employing other boiling fluids such as liquid metals.

It was clear that studies to determine the basic mechanisms which controlled the dynamics of two-phase coolants were very important. However, it was decided that the best initial approach to the problem consisted of combining both basic and applied developmental studies. The applied studies are very helpful in outlining the problem areas for reactor design purposes and can add much useful information to the understanding of the basic problems.

The initial objective of the present program is to promote coordination between the various investigators in this field of research in order to: 1) insure a rapid exchange of technical information by direct distribution of progress reports, 2) promote exchanges of technical ideas and philosophy by holding frequent informal working meetings, and 3) insure no unnecessary duplication of effort.

The present program is still in the early phases of development and is expanding at this time. The following studies that will be described are those now under way or proposed for initiation in the very near future, and as mentioned before, contain both basic and applied approaches to the problem of understanding two-phase flow phenomena.

1. *Argonne National Laboratory*—Investigations of many aspects of two-phase flow such as vapor-liquid slip, burnout, critical flow, phase distributions, and hydraulics with various fluids such as water, freon and sodium. These studies are performed as part of both the AEC Heat Transfer Program at Argonne and the work performed at the laboratory by participants of Associated Midwest Universities and directed by Argonne National Laboratory. This work was presented in more detail at the meeting by Mr. John Marchaterre.

2. *Oak Ridge National Laboratory*—Studies of high gravity (vortex) flow with boiling water, transition boiling regime studies, and heat transfer and fluid dynamics investigations of boiling potassium. This program was presented by Dr. Hoffman.

3. *General Electric Company - APED*—

a) Studies of two-phase flow frictional and contraction-expansion pressure losses with and without heat addition at high quality with vertical and horizontal flow.

b) Critical heat flux and two-phase flow in fuel rod bundles performed under the fuel cycle program.

c) Proposed studies of transition boiling (above burnout).

The above projects were presented by Dr. Levy.

4. *Dynatech Corporation*—Studies of transition from slug to froth or fog flow and two-phase flow pressure losses in venturis. These studies were presented by Dr. Moissis.

5. *Massachusetts Institute of Technology*—Investigations of condensation of liquid metal vapors, effects of surface character on boiling of liquid metals, film boiling in forced convection inside tubes, effect of electrostatic field on boiling, and studies of slug flow. These projects were presented by Dr. Warren Rohsenow.

6. *North Carolina State College*—Two-phase flow frictional and contraction-expansion pressure drop studies with boiling freon and water. This study was presented by Dr. Ferrell.

7. *Dartmouth College*—Investigations of the basic mechanisms of two-phase flow and boiling heat transfer. This study was presented by Dr. Graham Wallis.

8. *United Nuclear Corporation*—Studies of Two-Phase Flow Patterns under heat addition. This project is supported under the Joint U.S.-Euratom R & D Program.

9. *General Electric Company - General Engineering Laboratory*—Proposed studies of the stability of two-phase flow systems with change of phase. The principal investigator will be Dr. Novak Zuber.

10. *Allis-Chalmers*—Proposed investigations of void fractions and vapor-liquid slip ratio in large channels and with low liquid velocities.

11. *Babcock & Wilcox*—Proposed studies of the effect on critical heat flux of variable heat generation rates in both the axial and radial direction.

In addition to the above studies, other work is under way with AEC support which is not a direct part of the two-phase flow program, but supply useful information to the program and whose investigators attend the working meetings and participate in the report exchanges. These organizations are:

1. Hanford Laboratories (Two-Phase Flow Pressure Drops and Critical Flow Phenomena), Dr. J. Batch.

2. Westinghouse Corporation (Boiling Heat Transfer Studies), Dr. L. Tong.

3. Brookhaven National Laboratory (Boiling Potassium Studies), Dr. O.E. Dwyer.

4. University of Minnesota (Critical Flow Studies and Heat Transfer and Fluid Flow in Non-Circular Geometries), Dr. H. Isben and Dr. E.R.G. Eckert.

5. Aerojet-General Nucleonics (Boiling Rubidium and Cesium Program), Dr. D. Cochran.

6. Geoscience Ltd. (Vortex Flow Investigations with Boiling Mercury and Potassium), Dr. H. Poppendiek.

In addition to the above AEC contractors, the Lewis Laboratory of the National Aeronautics and Space Administration will be participating in the AEC Two-Phase Flow Program.

The aforementioned studies represent the present program. As to the future program, it is difficult at this time to predict exactly what work will be performed, since these details depend to a great degree upon the results of the present work. However, it is highly probable that more work in the basic areas of flow regime and phase distribution; stability, burnout heat flux, and inpile heat transfer investigations will be initiated in the not too distant future.

Once a more basic understanding of the phenomena of two-phase flow is obtained, it is expected that the studies in this program will become more applied in order to meet our primary objective of developing design correlations, based upon coordinated theoretical and experimental studies, for the prediction of burnout or critical heat flux, heat transfer rates, pressure drops, void fractions, and hydraulic instability in boiling reactor systems. This information will be necessary in order to mitigate the necessity for detailed heat transfer experiments to produce design data for every reactor project employing two-phase coolants, and to more nearly optimize these systems.

The above information was concerned with the Commission's two-phase flow program. However, it may be of interest to mention briefly some of the other heat transfer programs presently under way in the AEC's Office of Nuclear Technology.

1. *High Temperature Liquid Metal Studies*—Studies of high temperature liquid metals for use as reactor coolants and working fluids in advanced compact power systems. Investigations include various studies of the heat transfer and fluid flow characteristics of sub-cooled, boiling, and condensing mercury, sodium, potassium, rubidium, cesium, and lithium.

2. *High Temperature Gas Studies*—Heat Transfer, fluid dynamics, and component development studies with high temperature gases (to 4500°F) for use in space propulsion and high temperature process heat systems.

3. *Advanced Coolant Technology*—Investigations of improved coolants, such as present work on gas-solids suspensions for increased performance, reduced size, and cost of gas-cooled reactors.

4. *Methods to Increase Heat Transfer Rates*—Methods are continually being studied in order to increase over-all thermal efficiencies and power densities of reactor systems. Present methods being investigated include: electrical impulses, vibrations, and high gravity (vortex) flow.

5. *General Heat Transfer and Fluid Flow Studies*—Other projects under way include investigations of: a) heat transfer and fluid flow in fluidized beds and packed beds; b) temperature and velocity distribution in non-circular channels; c) determination of thermal contact coefficients between fuels and claddings; and d) fluid dynamics of multicomponent gas systems.

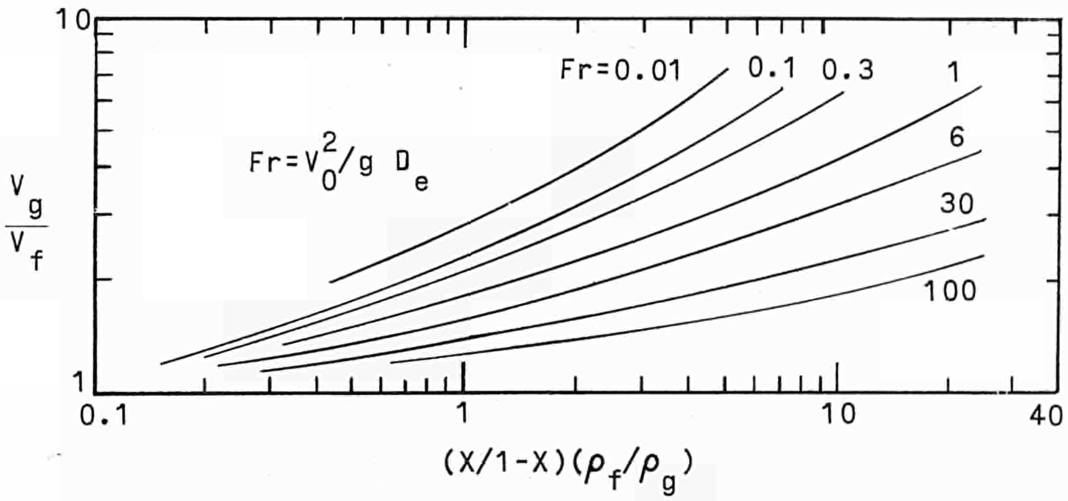
PRESENTATION OF THE TWO-PHASE FLOW PROGRAM AT ARGONNE NATIONAL LABORATORY

J. MARCHATERRE
Argonne National Laboratory, Argonne, Illinois, USA

The program in two-phase flow at Argonne National Laboratory can be divided into two parts.

The first part is that associated with the water reactor program and the second with the advanced systems and fast reactor safety programs. The emphasis in the boiling reactor system has been primarily on predicting the steady state behavior of boiling water reactors of the EBWR type.

An example of the type of studies that have been done are the studies aimed at predicting the vapor volume fraction in boiling systems. These studies have been concluded on a general correlation in terms of the velocity ratio. We have chosen the velocity ratio for correlation, though this is a very arbitrary choice. The results are shown in figure 1.



VELOCITY RATIO VS. VOLUMETRIC FLOW RATE AND FROUDE NUMBER

Figure 1

Our feeling now is that more accurate determinations will depend upon a study of individual flow patterns. However, while these more general studies are being pursued, design data must be developed to enable reasonable estimates of performance to be made. We are continuing these studies in downflow systems. Preliminary results are presented in ANL-6581.

Another aspect of boiling water reactor performance which is receiving study is the problem of the entrainment of the vapor in the downcomer in a reactor of the EBWR type. This problem has been studied in some detail, both analytically and experimentally.

The study of vapor carryunder and associated problems has been completed. A simple model for the carryunder phenomenon is postulated and an analytical expression for the mixture quality ratio of the downcomer to riser X_D/X_R is derived. The dominating factor in this analysis is the definition of a specific area in the riser from which carryunder emanates. Data taken from an atmospheric air-water loop are compared with the predicted values for weight percent carryunder for the parameter range studied: mixture qualities $(0.2\times10^{-3} < x < 2.0\times10^{-3})$, and downcomer velocities $(1 \text{ ft/sec} < V_D < 2.5 \text{ ft/sec})$. The proposed model proved to be quite successful and good agreement between measured and calculated values are obtained.

A dimensional analysis of the pertinent parameters affecting carryunder was also made and a series of dimensionless groupings were derived. These groupings were then used to develop empirical correlations for predicting carryunder. The empirical correlation adequately represents a series of high-pressure data over the following parameter range: void fractions $(0.1 < \alpha_R < 0.5)$, downcomer velocity $(0.5 < V_D < 2.5 \text{ ft/sec})$, and pressure ($P = 600$, 1000, 1500 psi). The correlation is shown in figure 2.

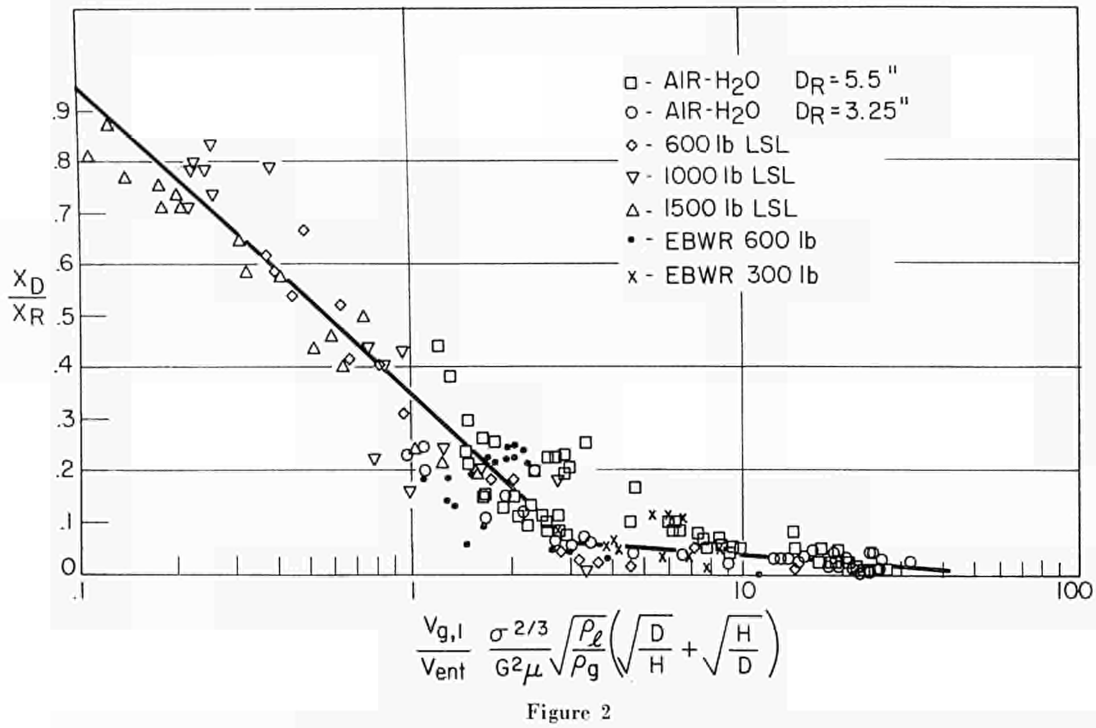


Figure 2

Data have also been obtained on the associated problem areas: namely, (1) bubble-size distribution $(0.01\times10^{-3} < x < 2.0\times10^{-3}$, $0.9 \text{ ft/sec} < V_s < 2.8 \text{ ft/sec})$; (2) bubble-size versus bubble-velocity relationship; (3) phase distributions within a conduit $(1.15 < V_s < 9.25 \text{ ft/sec}, 0.0004 < x < 0.004)$; and (4) downflow slip ratios ($P = 600$, 1000, 1500 psi; $0.003 < x < 0.12$; 19 ft/sec $< V_s <$ 4.5 ft/sec). The results of this study are summarized in ANL-6581.

These studies are being correlated with measurements being made on EBWR. The reactor core has been instrumented to measure recirculation flow rate steam value frac-

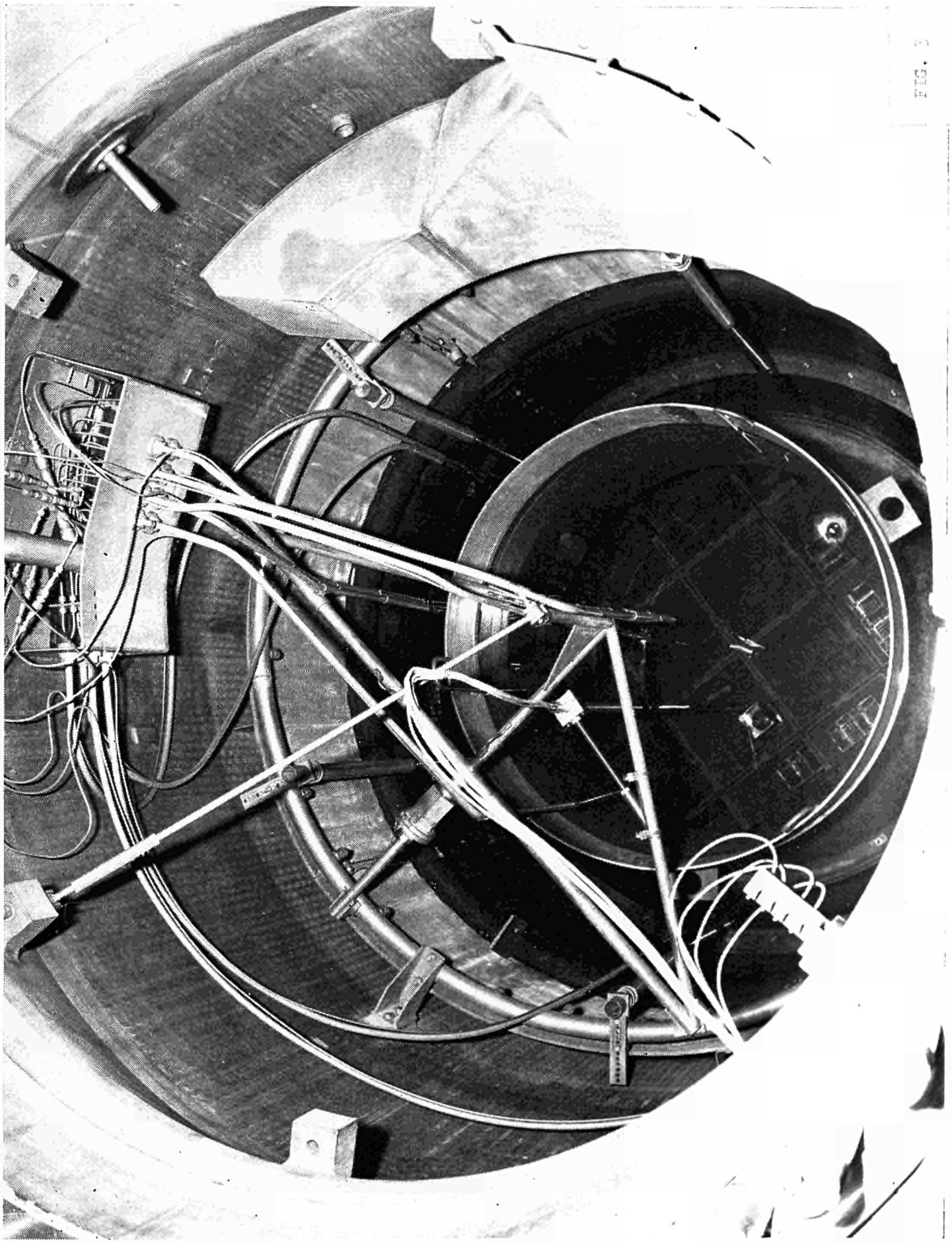


Figure 3

tions, and the amount of carryunder. Figure 3 shows some of the instruments in place in the core.

Another problem in two-phase flow that is under investigation is the critical flow of steam-water mixtrues. The maximum discharge of steam-water mixtures from conduits and breaks in vessels and pipes are subjects of considerable concern in the evaluation of nuclear reactor accidents and containment.

Much experimental work regarding the critical phenomenon in circular flow passages of considerable lengths (most cases greater than 10 inches) has appeared recently. Experimental data of basic value beyond about 360 psia (exit critical pressure) are fragmentary. Critical flows in straight tubes of considerable lengths differ significantly from the flow from a sudden break in a vessel containing highly pressurized saturated water. A sudden break in a vessel is a more realistic picture of what might happen in a nuclear reactor assembly in case of an accident. The dependence of critical flow rate from different shaped flow passages is at present unknown. Perhaps the most important question to clarify is the nature of the flow when a sudden break occurs in the cooling system of a reactor. Is the fluid leaving the break in physical equilibrium or in a metastable state ? This question is very important, inasmuch as the flow rate is very sensitive to the compressibility of the fluid.

A good way to explore these various problems is to use flow passages of varying lengths (both circular and rectangular), decreasing to a break at the vessel wall. This approach provides the opportunity to obtain data at pressures higher than 360 psia, and to carry out void fraction measurements under critical steam-water flows. The knowledge so obtained could be used to verify or disprove existing theories on critical flow, and could provide the basis for testing new analyses.

Preliminary experimental studies with Freon 11 have been carried out. The experiments (120 tests in total) covered a range of modified cavitation numbers between 1 and 500 length, diameter ratio of small diameter tubes between 2 and 55, and sharp-edge apertures of 9 different geometries.

It was found that below the modified cavitation number of 10 the fluid exhibits completely metastable single-phase flow. When the modified cavitation number exceeds 14, two-phase critical flow seems likely to occur. In the range of modified cavitation numbers between 10 and 14, unstable transitional flow occurs.

The Euler number may be correlated with modified cavitation number and length-diameter ratio. Prediction of discharge rates can be obtained from the correlations. Euler numbers for the apertures of various configurations including square, rectangular and eye-shaped were found to be in the same order of magnitude as those of circular shapes. The triangular and W-shaped orifices were found to possess a lower Euler number than circular ones.

$$\text{Cavitation } N_o = \sigma_c = [|P - P_V|/\rho]\ |V^2/Z_{gc}|$$

$$\text{Euler } N_o = g_c D\ (-dP/de)/\rho V^2$$

This work will now be extended to high pressure steam-water systems (up to 140 atm).

A number of studies of the transient behavior of two-phase systems are underway. One of these problems is the problem of oscillation in natural circulation systems. A large amount of experimental data on the transient behavior of these systems has been obtained. Since one method of studying a time-varying system is to perturb it externally, observing the changes in the variables, studies of the power-to-void transfer function have been made and a study of the flow-to-void transfer function is being planned. A

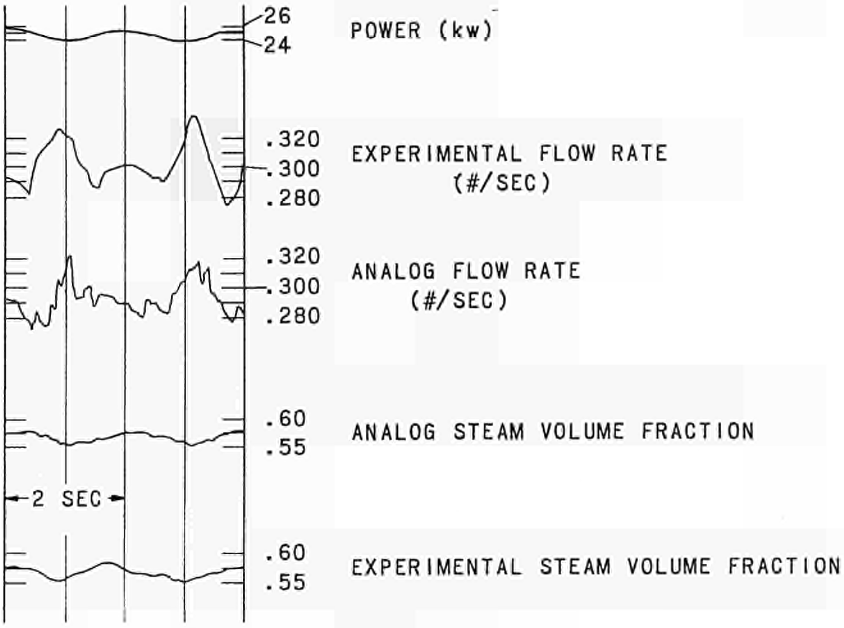


Figure 4
Transient comparison between analog model and experimental measurements

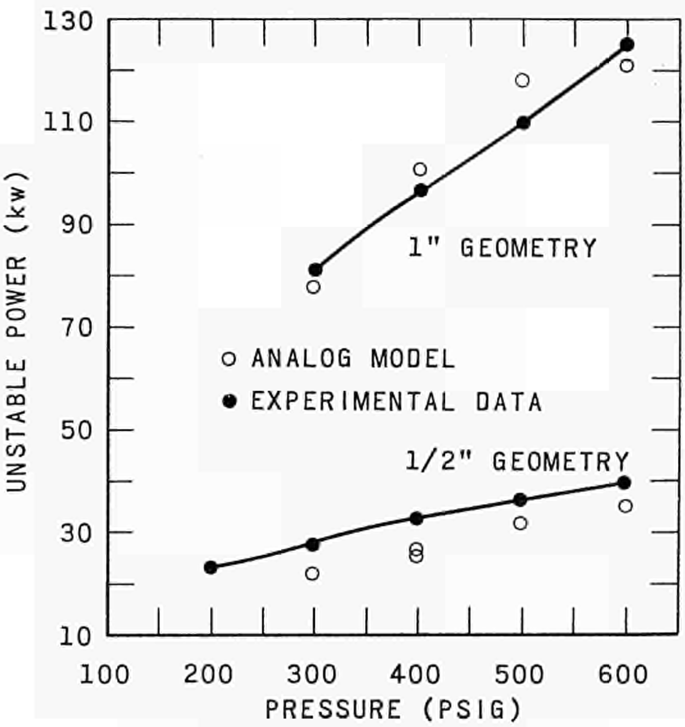


Figure 5
Comparison of predicted and measured power at the inception of oscillations

study is being made on a forced circulation system of the time-varying steam volume fraction in a test section caused by a sinusoidally-varying inlet mass velocity or test section heating power. The correlation of a dependent variable to the forcing variable provides a transfer function which can be compared to theoretical models of two-phase, time-dependent behavior in the two-phase portion of a circulating loop. The technique used to analyze the void fraction versus inlet flow compares the phase angle and amplitude ratio between the two signals electrically, using a noise-rejection cross-correlation of the signals.

Another approach is to write the time-dependent energy, continuity, and momentum equations and attempt a solution. This has been done and some of the results are shown in figures 4 and 5.

This type of study leads us to the second area of work being conducted at Argonne, namely, the area of fast reactor safety.

One of the major problems of nuclear safety in fast power reactors is related to core meltdown. There is a potential for large reactivity increase should meltdown lead to a significant increase in the effective fuel density. The calculation of this "reassembly is a complicated function of a great number of variables". One of the things which must be done is to predict coolant heat transfer and movement during transients.

Consequently, these studies are going to be extended to attempting to predict fluid movement during rapid exponential transients. The studies will be made first with H_2O in static systems, then extended to flowing systems, and finally if possible to sodium systems.

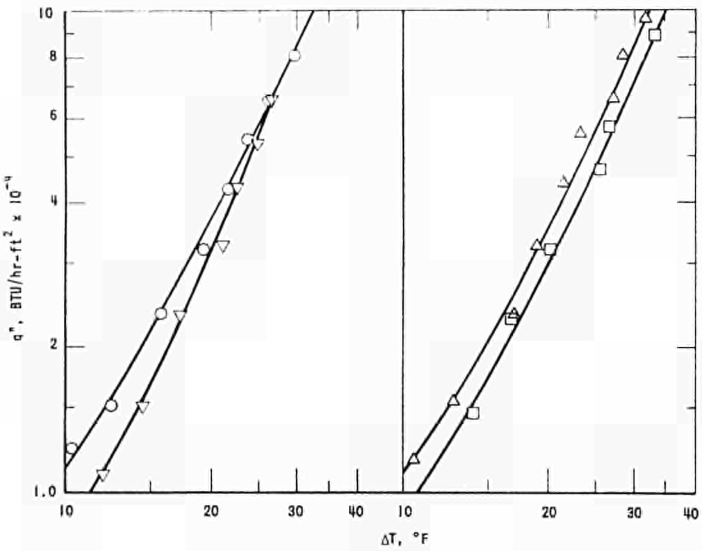


Figure 6

Heat flux as a function of temperature difference for water boiling from a mercury surface

1 ○ Clean mercury - Increasing heat flux

2 ▽ Used mercury - After series of runs in 1 with decreasing heat flux

3 △ Same as 1

4 □ Used mercury - After 6 hours of boiling, followed by 3 days of lying under liquid with increasing heat flux

Another problem that was studied in connection with fast reactor safety was the study of boiling from a liquid-liquid interface to attempt to gain some insight into the heat transfer mechanism when molten fuel was expelled into the coolant stream. Some of the results are shown in figure 6.

The fact that the interface may never be clean unless other precautions are taken raises the most serious question concerning these results. Considerable more work remains to be done in this area.

The last thing I will discuss is the boiling sodium experiments. One of the needs of the models we have previously used for predicting transient behavior is the steady state vapor volume fractions and frictional characteristics. A loop has been constructed to measure these in the low quality region.

One of the instruments that has been developed is the use of a electromagnetic flowmeter to measure vapor volume fractions.

Figure 7 shows the results obtained with a NaK-Argon loop and shows the vapor volume fractions measured by gamma attenuation and the electromagnetic flowmeter method compared to the results of the correlation shown in figure 1.

α_1	α_2	α_3
0.191	0.270	0.209
0.372	0.367	0.367
0.517	0.436	0.446
0.662	0.488	0.559
0.643	0.534	0.578
0.206	0.225	0.236
0.362	0.335	0.350
0.478	0.401	0.408
0.533	0.459	0.436
0.575	0.481	0.458

Legend

α_1 attenuation method
α_2 froude no. correlation
α_3 measured (E.M. flowmeter)

VAPOR FRACTION COMPARISON

Figure 7

STUDIES IN TWO-PHASE FLOW AND BOILING HEAT TRANSFER AT THE OAK RIDGE NATIONAL LABORATORY

H.W. HOFFMAN
Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

The Engineering Science Group of the Reactor Division at the Oak Ridge National Laboratory has been engaged for a number of years in a general program of experimental research in heat transfer and fluid mechanics directed to increasing the heat-removal capabilities of nuclear reactors. Studies have ranged from experiments into the influence and nature of added surface roughness on the rate of heat transfer to considerations of fundamental boundary-layer and bulk-flow dynamics, from investigations of multiphase and nonaxial flows in circular and noncircular geometries to evaluation of unusual coolants. The portion of this program relating to two-phase flow and boiling heat transfer is listed in summary in Table 1. Several of these studies are considered in detail in the remainder of this report.

Table 1

TWO-PHASE FLOW AND BOILING AND CONDENSING HEAT-TRANSFER STUDIES AT THE OAK RIDGE NATIONAL LABORATORY

Boiling Heat Transfer	
a. Water	— swirl flow with inlet generators and twisted tapes under subcooled and saturated conditions
	— natural circulation and forced convection in narrow parallel-plate channels under subcooled and saturated conditions
	— pool boiling inside horizontal tubes
	— effect of surface condition (time) on inherent variation of critical heat flux
	— pool boiling under agravic conditions
b. Ethylene glycol	— pool, axial, and swirl flow boiling
c. Freon	— stability and flow regimes for saturation boiling in seven-rod clusters
d. Liquid metals	— forced-convection axial and swirl flow boiling in vertical and horizontal tubes with saturated potassium
	— swirl-flow forced convection with subcooled mercury
	— boiling stability with saturation boiling of potassium
	— potassium boiling with flow parallel to rod clusters, in annuli, and within multiple tubes
e. Slurries	— pool and forced-convection boiling of ThO_2 slurries on horizontal tubes and vertical and horizontal strips under subcooled and saturated conditions
f. General	— prediction of critical heat-flux limits with flowing, subcooled, wetting liquids

Condensing Heat Transfer

a. Water	— condensation of steam in circular tubes of both uniform and tapered extent
b. Liquid metals	— condensation of potassium in horizontal and vertical (with up and down flow) tubes

Two-Phase Flows without Heat Transfer

a. Air-water	— definition of flow regimes in tubes and rod clusters under vertical and horizontal aspects
	— gas-liquid separation
b. Steam-water	— pressure drop in rod cluster geometries
c. Potassium vapor-liquid	— pressure drop in tubes and rod clusters
	— vapor-liquid separation

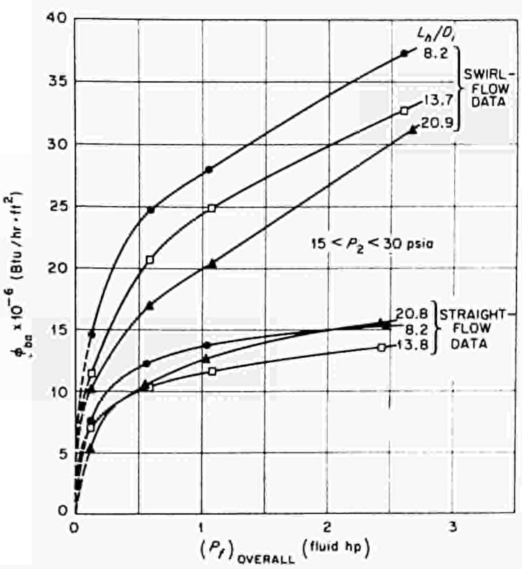
General

a. Water	— surface temperature transients with forced-convection under boiling and nonboiling conditions

---

Boiling with Swirl Flow. Phenomenological considerations of a swirl flow suggested that a significant increase in the magnitude of the critical (or "burnout") heat flux should occur for a boiling heat-transfer situation. It was reasoned that (1) the large radial pressure gradient associated with the rotating fluid would accelerate the movement of vapor away from the heat-generating surface, (2) the high pressure at the wall would reduce the size of the vapor bubbles, and (3) the higher liquid convection velocities at the wall [for a total mass flow equal to that for an axial flow through a tube of the same diameter] would postpone the initiation of boiling. Since the details of this work have been amply reported in the technical literature (a list of references is attached [1-4]), the following discussion merely highlights some of the more pertinent results.

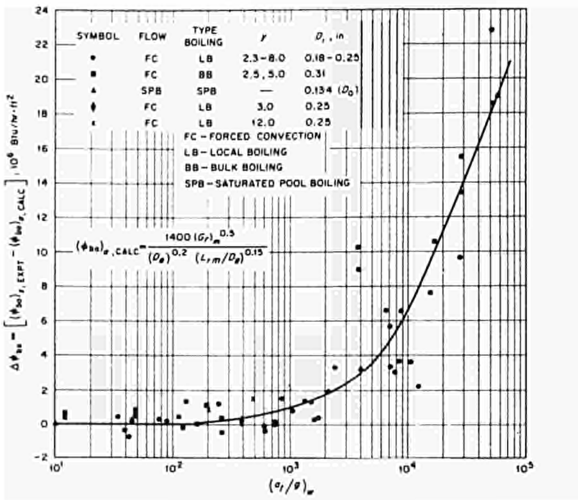
As anticipated, the critical heat fluxes realized with swirling subcooled water were much greater than had been previously observed with axial flows. Thus, in short tubes with tangential inlet vortex generators, ϕ_{crit} was observed to be as high as



Comparison of Swirl and Axial Burnout Heat Fluxes
in Identical Tubes at Constant Pumping Power.
Figure 1

55×10^6 Btu/hr·ft² and in longer tubes with full-length helical (twisted) tapes, the critical flux was as large as 37×10^6 Btu/hr·ft². Typical swirl and axial flow results are compared at equivalent pumping power (P_f = total pressure loss including end effects $\times$ the volumetric flow rate) in figure 1; an improvement due to swirl flow of approximately a factor of two is noted. An alternative representation of these results is given in figure 2 in the form of the incremental increase in the critical heat flux due to the action of the swirl flow alone. The axial flow contribution was calculated using a modified form of the correlation of Lowdermilk *et al.* [5] with zero exit subcooling; the swirl flow results appear to be independent of the extent of exit subcooling. It is observed that for values of the a/g parameter (local acceleration/normal graitational force) below 200, the effect of swirl flow is negligible; by way of reference for a flow having a resultant axial velocity of 60 ft/sec within a 1/4-in.-ID tube containing a twisted tape of y = 2, the a/g value is 6.6×10^3.

UNCLASSIFIED
ORNL-LR-DWG. 75545

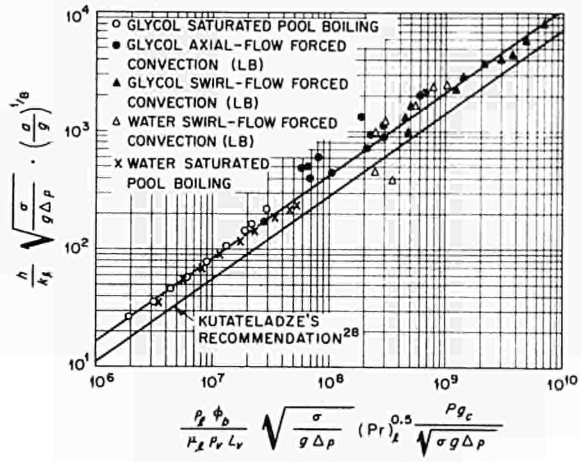


Variation of
Swirl-Flow Incremental Burnout Heat Flux
with Centrifugal Intensity.
Figure 2

Data for both pool boiling and axial and swirl forced-convection boiling with water and ethylene glycol have been combined using the parameters proposed by Kutateladze [6] with the empirical introduction of the factor, $(a/g)^{1/8}$, to bring the swirl-flow data into better agreement with the pool and axial-flow results. As seen in figure 3, these results fall approximately 50% above Kutateladze's recommended correlation.

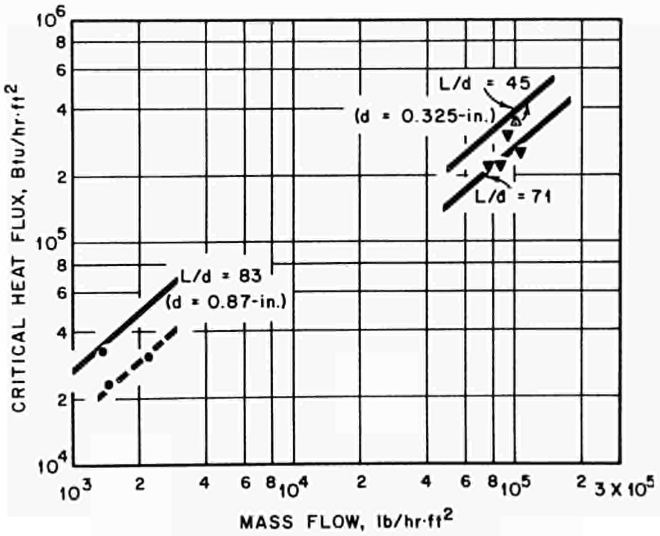
Forced-Convection Saturation Boiling with Liquid Potassium. Boiling liquid metals have been suggested as nuclear-reactor coolants and as a source of high-temperature vapors for power conversion systems. Since data on the heat-transfer characteristics of these fluids under forced-convection boiling conditions were not available and question existed as to the validity of direct extrapolation of boiling-water correlations to this situation, an experimental study of the forced-convection saturation boiling of potassium in a vertical tube was initiated.

An electrodynamic pump was used to circulate the liquid potassium through a closed loop consisting of a test boiler, a liquid-vapor separator, an aircooled condenser, and calibrated volumetric tanks for measuring the flows of the separated vapor and liquid streams so as to establish the quality of the boiler effluent mixture. Two type 347 stainless steel tubes of different diameters have been utilized as the boiler section: (1) a 6-ft long, 0.87-in.-ID (0.065-in.-wall thickness) tube heated by radiation from a set of six surrounding concentric electric-resistance elements and instrumented with 0.008-in.-



Combined Nucleate Boiling Heat-Transfer Data
for Ethylene Glycol and Water.
Figure 3

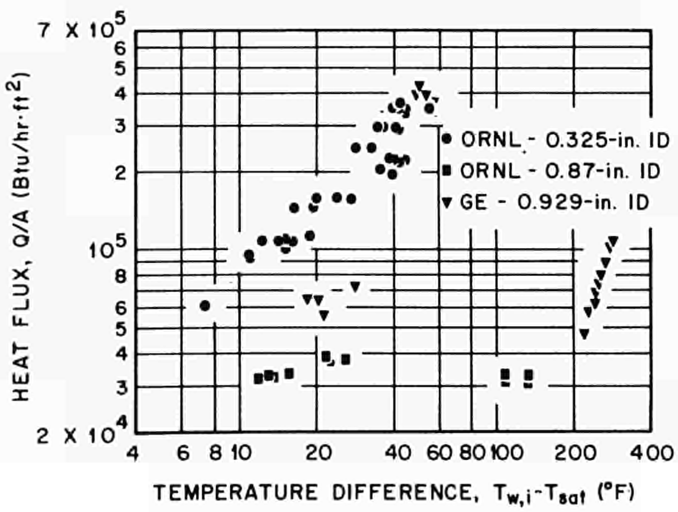
dia Pt-Pt 10% Rh thermocouples welded to the tube outside surface at the approximate mid-point of each heater, and (2) a 44-in. long, 0.325-in.-ID (0.028-in.-wall thickness) tube brazed within a segmented (21 equal sections) copper sleeve of 5-in. outside diameter, heated by conduction from a set of 21 external heaters, and instrumented with 0.040-in.-OD sheathed Chromel-Alumel thermocouples passing axially through and radially distributed within the individual copper segments. In the latter boiler section, the thermocouple readings in each segment were extrapolated to give the temperature at the boiling surface and, in combination with the copper conductivity, the heat flux through each segment.



Critical Heat Flux for Saturated Potassium
in a Vertical Tube.
Figure 4

Experiments thus far have been directed toward the accumulation of data on the critical heat flux, with information on two-phase pressure losses and heat-transfer coefficients being a secondary goal. The desired data were obtained by maintaining the heat flux at the boiling interface at some fixed value while decreasing the flow in a stepwise pattern until "burnout" occurred. The burnout point was defined by the automatic interruption of boiler power when the wall temperature climbed to some predetermined value. In the larger diameter boiler, burnout was preceded by wall-temperature oscillations of $\pm$ 30 to $\pm$ 50°F (and associated flow and pressure fluctuations) whose maxima eventually exceeded the 1700°F limit. For the smaller diameter boiler, the burnout condition was evidenced by a sharp rise in wall temperature with an equally sudden reduction in the over-all pressure drop; a wall temperature rise of 50°F was taken as the power cutoff criterion.

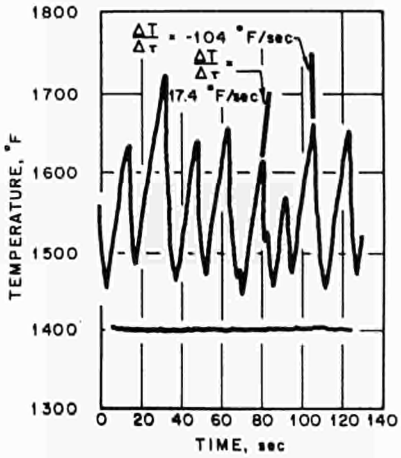
The experimental results obtained for the critical heat flux are shown in figure 4 in comparison with the dimensional correlation of Lowdermilk *et al* [5] for the saturation boiling of water in a vertical tube under the conditions of low inlet liquid velocity and high exit quality (70-90% for the data shown). It is interesting to note the reasonable agreement of the water and potassium data.



Heat Transfer with Saturated Potassium Boiling
in a Vertical Tube.
Figure 5

Results for heat transfer below the critical heat flux are given in figure 5 as a function of the wall superheat; the data indicated by the circles were obtained in the small-tube boiler, by the squares in the large-tube boiler. The scatter in the temperature difference, $T_w - T_{sat}$, at any given flux level can be associated with a systematic variation in exit quality; i.e., within each group of data points, the quality generally increased with decreasing ΔT. The inverted triangles are the results of Brooks [7] obtained in a double-tube boiler, sodium to potassium; the data at high ΔT have been described as corresponding to film boiling. By way of further comparison, the curve for saturation pool boiling with water roughly agrees with these results, having a maximum at $q/A =$ 400,000 Btu/hr·ft² for $\Delta T = 40$°F and falling off on either side through the large diameter tube data.

It was observed during some phases of operation with the 0.87-in. tube that stable outside-tube-wall temperature oscillations of very high amplitude existed. Under such circumstances, burnout occurred at heat-flux values only 60% as large as those previously attained (dashed line, figure 4). The wall-temperature trace in figure 6 typifies this condition; comparison is made to a stable run at approximately the same inlet mass flow and net vapor generation (in excess of 60%). The frequency of the temperature fluctuations was of the order of 3 to 5 cycles/min with wall-temperature change rates



Outside-Tube-Wall Temperature during Period of Stable Oscillations with Saturated Potassium Boiling in an 0.87-in.-ID Vertical Tube.

Figure 6

of $+17°F/sec$ and $-104°F/sec$. This pattern suggests that film boiling alternated with nucleate boiling; the experimentally observed rate of rise of wall temperature is consistent with the value anticipated for the heat input involved if cooling of the inside-tube surface were suddenly discontinued. Again, extreme oscillations in boiler flow and pressure were noted; these may reflect a chug-flow condition due to flashing across a restriction at the boiler inlet or some phenomenon associated with a narrowing of the flow passage at the boiler exit.

REFERENCES

[1] W.R. GAMBILL, N.D. GREENE, "Boiling Burnout with Water in Vortex Flow," *Chem. Eng. Progr.* **54**, No. 10, 68-76 (1958).

[2] W.R. GAMBILL, R.D. BUNDY, R.W. WANSBROUGH, "Heat Transfer, Burnout, and Pressure Drop for Water in Swirl Flow through Tubes with Internal Twisted Tapes," *Chem. Eng. Progr. Symposium Series* **57**, No. 32, 127-137 (1961).

[3] W.R. GAMBILL, R.D. BUNDY, "An Evaluation of the Present Status of Swirl-Flow Heat Transfer," *ASME Paper* No. 62-HT-42 (August 1962) [copies available from ASME until June 1, 1963].

[4] W.R. GAMBILL, R.D. BUNDY, "High-Flux Heat-Transfer Characteristics of Pure Ethylene Glycol in Axial and Swirl Flow," *AIChE Journal* (January 1963) [to be published].

[5] W.H. LOWDERMILK, C.D. LANZO, and B.L. SIEGEL, "Investigation of Boiling Burnout and Flow Stability for Water Flowing in Tubes," *Nat. Adv. Comm. Aero.*, TN-4382 (September 1958) [available through Nat. Aero. and Space Adm., Washington, D.C.].

[6] S.S. KUTATELADZE, *Heat Transfer in Condensation and Boiling*, 2nd ed., Chapter 11, Mascow-Leningrad, 1949, 1952. AEC Transl. 3770, August 1959.

[7] R.D. BROOKS, "Alkali Metals Boiling and Condensing Investigations," *Quar. Prog. Rep. Jan. 1—Mar. 31, 1962,* General Electric Co., Flight Propulsion Laboratory, Cincinnati, Ohio.

TWO-PHASE FLOW AND HEAT TRANSFER STUDIES AT GENERAL ELECTRIC COMPANY

S. LEVY, E. JANSSEN, E.E. POLOMIK, E.P. QUINN and F.E. TIPPETS

General Electric Company, Atomic Power Equipment Department
San Jose, California, USA

The Atomic Power Equipment Department (APED) of General Electric Company has been engaged in extensive studies of two-phase flow and heat transfer. These studies cover investigations of two-phase pressure drop, critical heat flux or "burnout", vapor fractions, hydraulic stability, and heat transfer to superheated steam. Tests and analyses of an applied and basic nature have been performed in all the above fields. It is the purpose of this report to summarize some of the latest results obtained at APED. For the sake of brevity, the following remarks will be confined to recent findings in the field of two-phase pressure drop and burnout heat transfer. A brief description will also be given of a proposed investigation in transition boiling.

TWO-PHASE PRESSURE DROP*

The most important investigations at APED in two-phase pressure drop are presently being carried out under a contract for the U.S. Atomic Energy Commission. These studies can be broken down into three groups. The first group is concerned with two-phase flow of high pressure (600 to 1400 psia) steam-water mixtures in rectangular and circular channels. The tests include measurements of two-phase pressure drop, void fraction, and flow structure for upwards, downwards, and horizontal flow. The second group of investigations consists of similar studies in simplified contraction and expansion geometries. The final group of tests deals with two-phase pressure drop tests with and without heat addition to establish the effects of heating on two-phase flow.

The status of this work can be summarized as follows:

1. Extensive pressure drop data have been taken in rectangular channels at 600, 1000, and 1400 psia. High speed motion movies have also been made of the flow structure. The tests covered upwards, downwards, and horizontal flow directions.

2. Extensive pressure drop results and movies have been accumulated for the same conditions in simplified two-dimensional contraction-expansion geometries.

Additional work planned for the near future includes the measurement of voids for the geometries tested above and the accumulation of pressure drop data in circular

* Dr. E. Janssen has project responsibility for this work.

channels. Tests with and without heating are planned, and simplified circular contraction-expansion geometries are to be studied.

Typical pressure drop profiles measured in a 1/2-inch $\times$ 1-3/4-inch horizontal rectangular channel are plotted in figure 1. The pressure drop data are obtained from taps located along the flow direction on the 1/2-inch face of the channel. Examination of figure 1 reveals that the test results fall on a straight line a short distance beyond the entrance point for both the single phase and two-phase run. According to figure 1, the

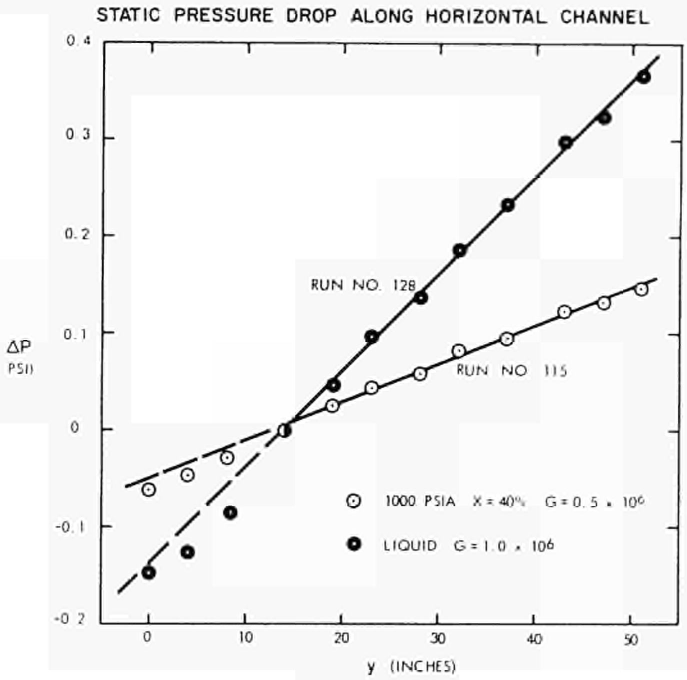


Figure 1

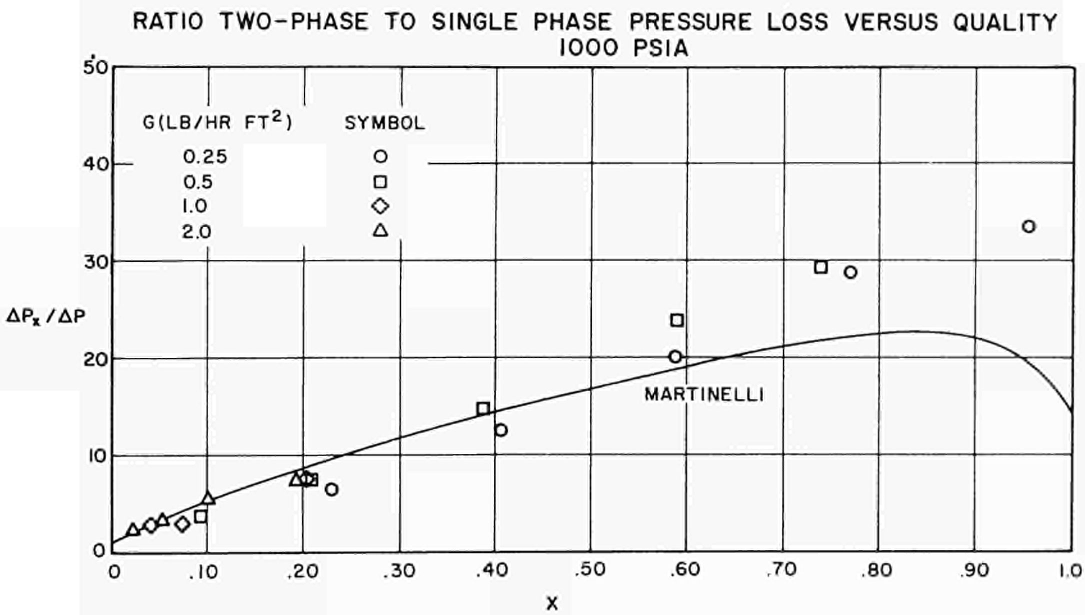


Figure 2

entrance length with high pressure steam-water flow is of the order of 20 hydraulic diameters and is of about the same magnitude as in single-phase flow.

The frictional pressure drop in single-phase and two-phase horizontal flow can be readily calculated from the slopes of the straight lines shown in figure 1. The ratio of two-phase to single-phase frictional pressure drop can next be obtained from the computed slopes. Values of this ratio at 1000 psia are plotted in figure 2. Comparison with the accepted Martinelli-Nelson [1] curve shows that the experimental data are slightly below the Martinelli-Nelson curves at low steam quality and well above it at higher steam contents.

Similar test data have been obtained for the channel oriented upwards and downwards. Measured pressure gradients for all three flow directions are plotted in figure 3. We can observe again that the flow reaches fully developed conditions a short distance from the inlet. It should be noted that in figure 3 the profiles shown for vertical up and downflow include both the frictional and hydrostatic head changes.

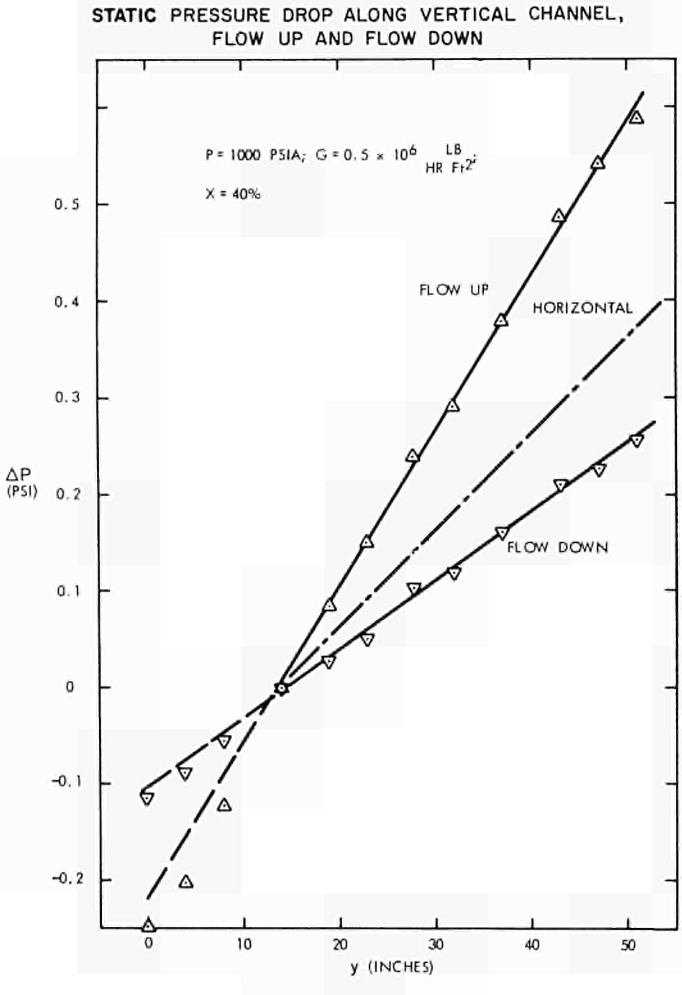


Figure 3

Since void measurements have not yet been made, it is not possible to separate the frictional and hydrostatic head losses in figure 3. Some preliminary trends can, however, be deduced by using the void data obtained by Larson [2] at 1000 psia. If we postulate that Larson's results are valid for all three flow directions, we can calculate the frictional pressure losses from the measured pressure profiles and we can once again calculate the Martinelli-Nelson multiplier. The ratio of two-phase to single-phase frictional pressure drop is plotted in figure 4. It can be seen from figure 4 that the calculated ratios are in relatively good agreement for all three directions at high flow velocity (which tends to substantiate the assumption of equal voids). At low flow rates, the calculated ratios become more dependent upon flow direction and in particular at low steam qualities the deviations become large. Another interesting conclusion can be drawn from figure 4. The high flow results tend to fall below the Martinelli-Nelson curve and appear to be lower than the results obtained at lower flows. A similar decrease in the Martinelli-Nelson multiplier with increased flow has been reported by other investigators for high pressure steam-water mixtures [3],[4]. Additional test runs and actual void data are to be obtained in the near future to confirm the preliminary results shown in figure 4.

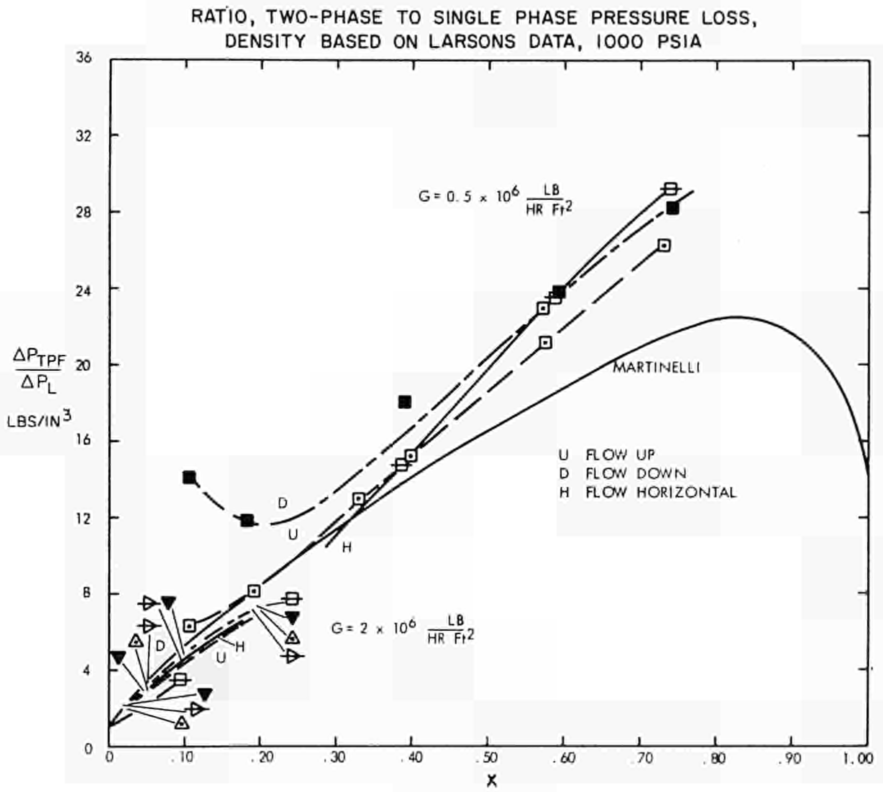


Figure 4

Extensive single-phase and two-phase pressure drop data have been obtained in two-dimensional contraction-expansion geometries. The test results have not been fully reduced and will not be described at this time.

Some high speed motion pictures have also been taken at 1000 psia in a rectangular channel with and without contraction-expansion geometries. Pertinent observations derived from projecting the movies are summarized below:

1. Stratification exists in a rectangular channel in horizontal flow. At 1000 psia and a flow rate of 10^6 lb/hr-ft^2 the stratification is strong at a steam quality of 5 per cent, decreases as the steam content is raised and practically disappears at 20 per cent

steam qualities. Water films can be discerned at the top and bottom surface of the channel, and the water film on the bottom surface appears thicker than on the top surface.

2. The annular pattern is dominant in a vertical rectangular channel at 1000 psia. Water films may be seen on both sides of the channel and the thickness of the film decreases as the steam content is raised at constant flow rate.

3. The flow structure at 1000 psia in a 1/2-inch vertical channel that contracts to 0.200 inch for 0.100 inch and expands again to 1/2 inch is as follows: an annular pattern exists ahead of the contraction and persists into the inlet of contraction; beyond the contraction, the flow is homogenized and takes the form of a jet with strong mixing along the jet boundaries, some flow reversal occurs part of the time beyond the contraction and the flow is unsymmetrical and time varying.

CRITICAL HEAT FLUX OR "BURNOUT" INVESTIGATIONS

Four important investigations have been recently completed at APED in this area. They are:

1. Critical heat flux determination in annular geometries heated on the inside surface only.

2. Critical heat flux tests in a 9-rod geometry.

3. Observations of the occurrence of critical heat flux conditions at high pressure in a rectangular channel.

4. Study of methods of increasing the critical heat flux.

A summary of the results obtained in each of these investigations is presented below.

*Annular Geometry Test Results**

Approximately five hundred test points have been taken in an annular geometry heated on the inside surface only. The test results cover the following range of conditions

Rod diameter:	0.375 to 0.540 inch
Hydraulic diameter:	0.245 to 0.500 inch
Heated length:	29 to 108 inches
Pressure:	600 to 1450 psia

Typical results are illustrated in figures 5, 6, and 7. Figure 5 shows that a change in heated length from 9 to 6 feet does not influence the critical heat flux. The effect of heated rod diameter is also negligible as inferred from figure 6. Test data obtained at Columbia University for a large heated rod of 1.375 inch are plotted on the same figure to confirm this trend. Figure 7 illustrates the effects of changing the hydraulic diameter from 0.335 to 0.500 inch. Within the range shown on figure 7, the role of hydraulic diameter appears negligible. Additional tests performed at various hydraulic diameters, however, show that the critical heat flux passes through a maximum as the

* Dr. E. Janssens had the principal responsibility for this work.

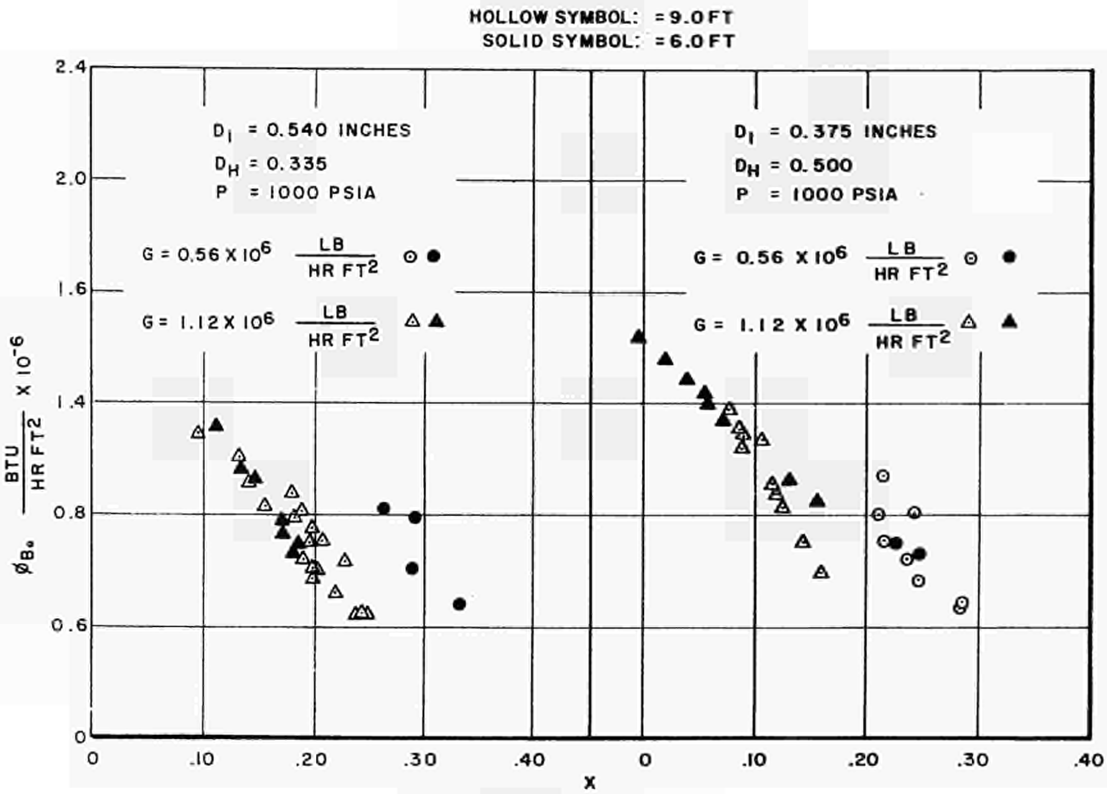


Figure 5
Effect of heated length

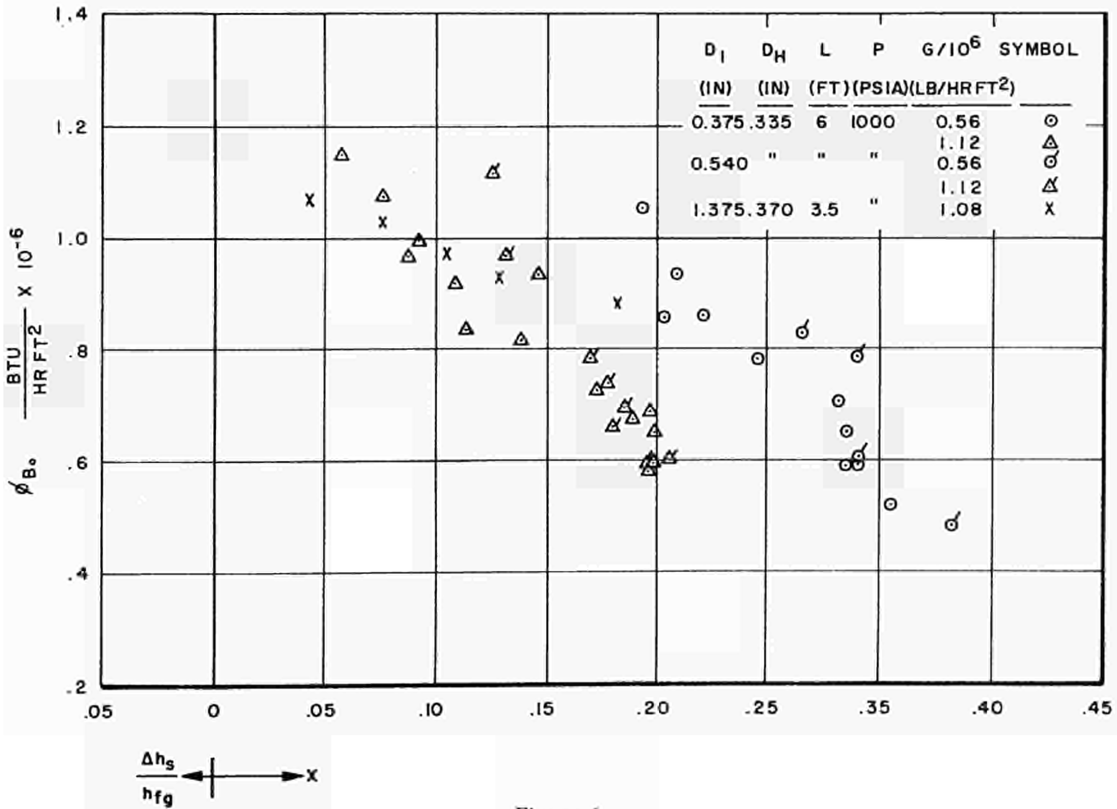


Figure 6
Effect of heated rod diameter

hydraulic diameter is varied. The critical heat flux decreases at both low and high hydraulic diameters and is the highest in the region of about 0.250 inch.

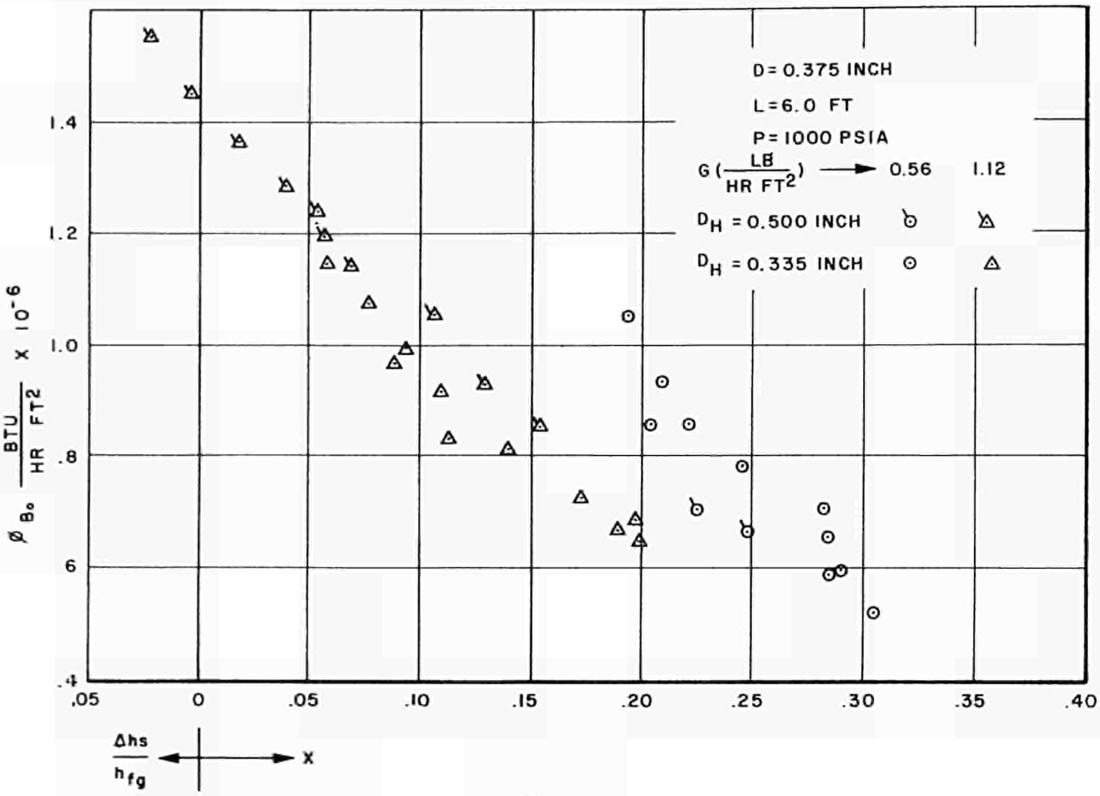


Figure 7
Effect of hydraulic diameter

The effect of flow rate can also be noted from figures 5, 6, and 7. The critical heat flux decreases as the mass flow rate is increased. This is clearly visible in all three figures. It should also be observed that the critical heat flux does not exhibit a maximum at low steam qualities as reported by other investigators [5]. This may be due to the fact that all test data were taken with subcooled water at the inlet of the test section. Another important result obtained from the tests is the effects of system pressure. The critical heat flux was found to increase as the pressure was reduced from 1450 to 600 psia.

Application of the extensive test results to reactor design is discussed in reference [6]. Design limit curves are developed based upon the annular data. These limit curves are next compared to data obtained with special geometries at APED and with data reported by other investigators.

*Multirod Test Results**

Multirod tests have been carried out in 3, 7, and 9-rod clusters. A schematic drawing of the 9-rod test section and its instrumentation is shown in figure 8. The three rods A, B, and C were instrumented and were so designed that they operated at a higher heat flux than the remaining rods. A steam water mixture instead of subcooled water

---

* E.P. Quinn and E.E. Polomik had principal responsibility for 9-rod work reported here.

was used at the inlet of this test section. Some typical test results obtained at 1000 psia are shown in figure 9 [7]. Plotted in the same graph as dashed are the limit curves recommended in reference [6]. Also shown on the same figure as solid lines are the mean lines obtained for the annular test section heated on the inside only. Examination of figure 9 reveals that

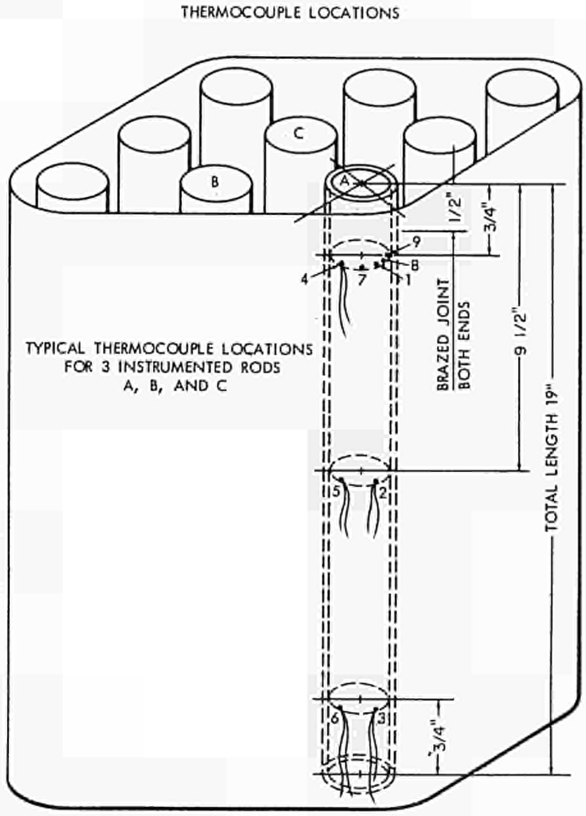


Figure 8

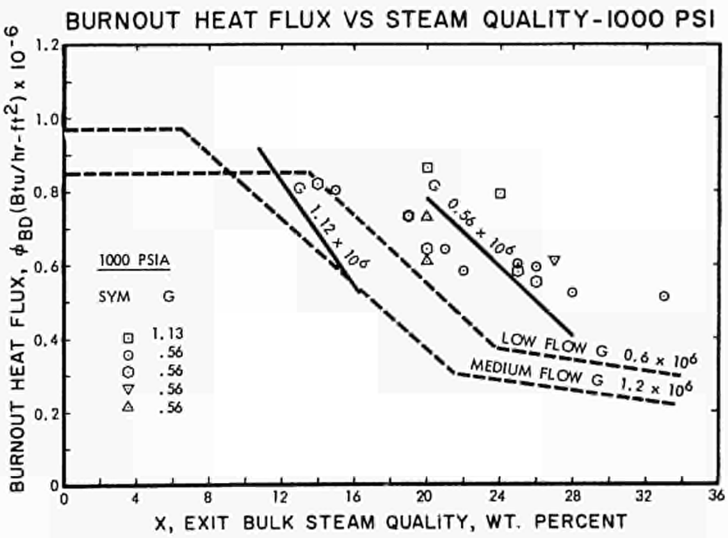


Figure 9

1. The multirod data fall above the limit lines. They are also within the range of the annular test results.

2. The effects of flow rate are not as clearcut as for the annular geometry. In fact, a reversal of the trend obtained in the annular tests appears to take place.

Thermocouples installed in the test section made it also possible to determine the location where the critical heat flux first develops. The critical heat flux was found to always occur in rods A and B and in the portions of these rods facing the rectangular channel.

*Observations of Critical Heat Flux Conditions***

High speed motion pictures of boiling water flow patterns have been taken in a rectangular channel 0.5×2.10 inch at 1000 psia [8]. The movies were taken over a range of coolant flow rates and coolant enthalpy values. Observations were made prior to and at the critical heat flux conditions. The motion pictures yield substantial evidence that a wavy turbulent liquid film flows against the heated surface and that the film becomes more and more agitated as the critical heat flux is approached.

An analytical model was postulated based upon these observations and checked against the 80 critical heat flux points obtained in the rectangular channel. The model was also compared with other available test results.

Study of Methods to Increase the Critical Heat Flux

One method successfully tested was the use of surface roughness on unheated surfaces. A roughened liner, as shown in figures 10A and B, gave substantial improvements in the critical heat flux. It is postulated that the surface roughness minimizes the accumulation of water on the unheated surface, thus increasing the amount of water available to the heated rod. Tests have been performed with different roughness size and spacing. The tests show a substantial gain, especially in the high steam quality region.

TRANSITION BOILING

From a design viewpoint, one is not only interested in the conditions at which the critical heat flux occurs, but also in what happens beyond it. Some recently completed tests [9] have shown that operation beyond the critical heat flux is possible and that such operation is accompanied by temperature oscillation in the transition boiling region.

A development program has been submitted to the Joint USAEC-Euratom Board to investigate the heat transfer process in the transition region in forced convection flow. Two series of tests at high pressure have been proposed. The first group of tests consist of measuring the occurrence of critical heat flux and of operating beyond it with a multirod geometry. The second group of tests is concerned with the mechanism of heat transfer in the transition boiling. Observation of flow and detailed surface temperature maps are proposed to understand the cause and magnitude of the oscillations.

** Dr. F.E. Tippets was responsible for this work.

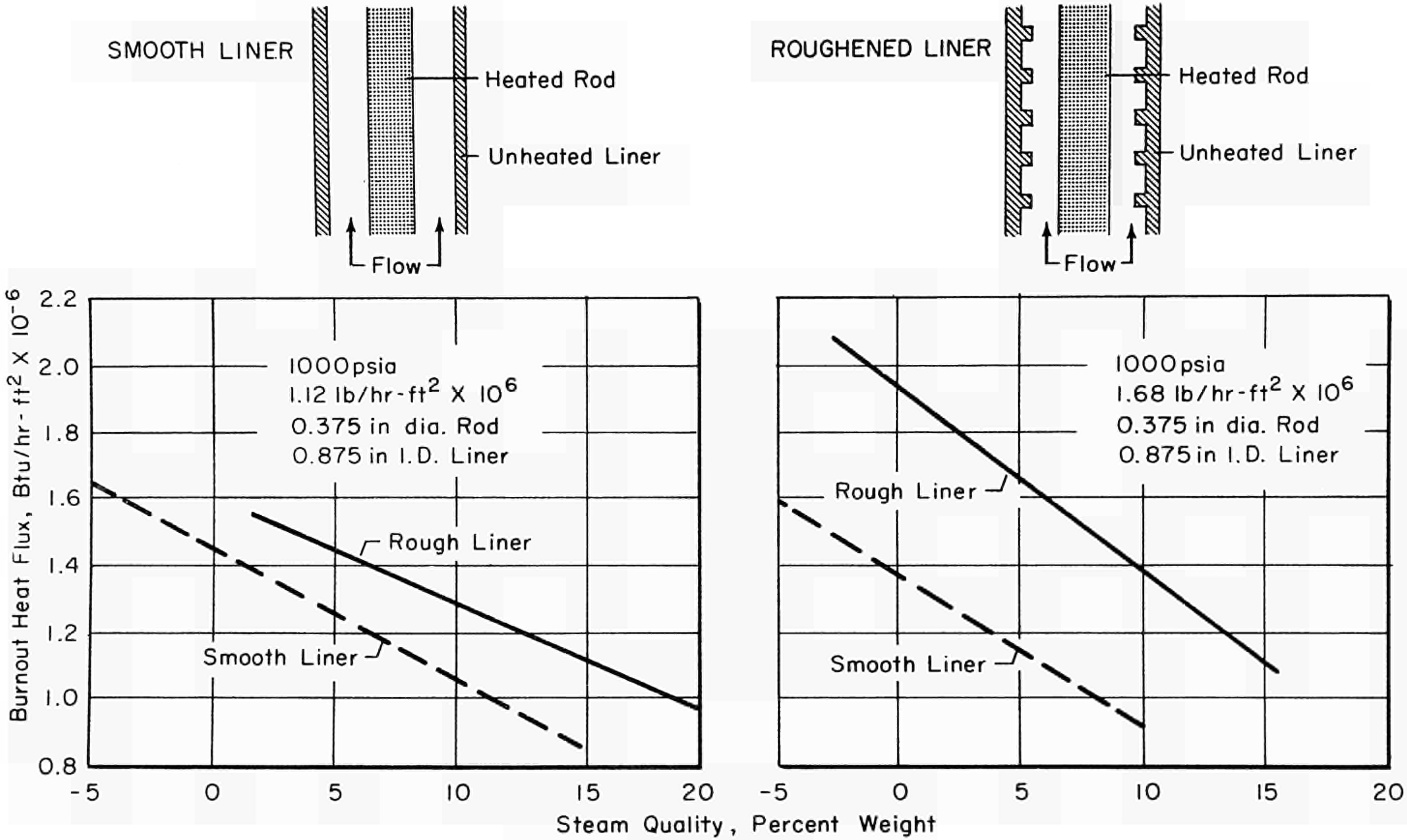


Figure 10 A & B
Increase in burnout heat flux due to roughened liner

REFERENCES

[1] R.C. MARTINELLI and D.B. NELSON, "Prediction of Pressure Drop During Forced Circulation Boiling of Water," Trans. ASME, Vol. 70, 1948, pp. 695.

[2] H.C. LARSON, "Void Fraction of Two-Phase Steam-Water Mixtures," M.S. Thesis, University of Minnesota, 1957.

[3] N.C. SHER and S.J. GREEN, "Boiling Pressure Drop in Thin Rectangular Channels," WAPD-477, 1958.

[4] R.H. MOEN, "An Investigation of the Steam-Water System at High Pressures and High Temperatures."

[5] M. SILVESTRI, "Two-Phase (Steam and Water) Flow and Heat Transfer," Inst. Heat Transfer Conference, Paper No. 39, Boulder, Colorado, 1961.

[6] E. JANSSEN and S. LEVY, "Burnout Limit Curves for Boiling Water Reactors," APED-3892, April 1962.

[7] E.E. POLOMIK and E.P. QUINN, "Multirod Burnout at High Pressure," GEAP-3940, 1962.

[8] F.E. TIPPETS, "Critical Heat Flux and Flow Pattern Characteristics of High Pressure Boiling Water in Forced Convection," GEAP-3766, 1962.

[9] E.E. POLOMIK, S. LEVY and S.G. SAWOCHKA, "Film Boiling of Steam-Water Mixtures in Annular Flow at 800, 1100, and 1400 psia," ASME Paper No. 62-WA-136.

TWO-PHASE FLOW STUDIES

R. MOISSIS
Massachusetts Institute of Technology, Cambridge, Massachusetts, USA

and

Dynatech Corporation, Cambridge, Massachusetts, USA

The work reported has been performed at Dynatech Corporation in connection with the development of a time-variant, self-controlling outlet steam quality device in boiling water nuclear reactors.

The control device consists of two venturi tubes one each at the inlet and outlet of a reactor flow channel. Since the mass flow rate through both venturis is the same, and assuming that the two phase mixture at the outlet venturi behaves as a homogeneous fluid, then the ratio of the pressure drops across the inlet and exit tubes is a measure of the flowing quality.

In the control device under consideration, the ratio of the two pressure drops is fed into an error comparator which weighs the existing ratio against a specified value. Deviations from the null position activate a flow control valve which regulates the inlet flow to the channel and thus maintains the exit quality at the required level.

The principle of the device continues to hold good even if the homogeneous flow assumption is not valid. What is essential is that the two phase flow characteristics of the system be predictable. For this reason some basic Two Phase Flow studies have been initiated.

The first topic of investigation is the study of pressure drop and discharge coefficients in a vertical venturi in two phase flow. An analytic model which takes into account slip velocity effects has been proposed. Visual observations of the flow reveal that various flow regime transitions may occur in the venturi. A bubbly-frothy mixture at the venturi inlet assumes a semi-annular configuration at the throat and returns to its original flow pattern in the diffuser section.

Experiments performed in an air-water system have been correlated successfully by means of the proposed model. The equations derived can be used for calculating pressure drops for volume qualities from zero to ninety percent. For volume qualities below forty-five percent, however, the homogeneous fluid assumption can be used with equal or slightly better accuracy.

Since flow regime transitions have been found to be of significance in predicting the performance of the system, a study of flow configuration changes has been undertaken in another phase of the reported project. In particular, an analysis has been presented which describes and predicts the process of transition from the non-homogeneous slug to the homogeneous frothy flow regime in the flow of two-phase mixtures in vertical pipes.

Although some idealizations have been incorporated in the proposed model, the approach is purely analytical and involves no empirical corrections or correlating constants. The main conclusions of the investigation are as follows:

1) The transition from slug flow to froth flow is due to a Helmholtz hydrodynamic instability of the liquid film-to bubble interface.

2) The transition depends on the size of bubbles in the slug flow. Since a given system contains bubbles of various lengths, the transition cannot be defined in terms of a line but can only be bounded within a band.

3) Increasing the pipe diameter or the Weber number accelerates the transition process.

The results of the theory compare favorably with existing visual flow regime studies.

SOME CURRENT AND FUTURE RESEARCH IN TWO-PHASE FLOW AT THE MASSACHUSETTS INSTITUTE OF TECHNOLOGY HEAT TRANSFER LABORATORY

W.M. ROHSENOW
Massachusetts Institute of Technology, Cambridge, Massachusetts, USA

CONDENSATION OF LIQUID METALS

Currently, apparatus is being constructed for condensation of mercury on the outside surface of a 7/8" diameter 6" long vertical bayonet nickel tube with cooling water inside. The thickness of the condensate film will be measured by the gamma ray attenuation—the source is a one millicurie Co 57 piece and the detector, a NaI crystal.

Many analyses of liquid metal condensation are in the literature. The very small amount of available data fall far below (factor of 2 to 5) the heat transfer coefficients predicted by analysis.

The objective of the present project is to attempt to discover why there exists such a large discrepancy between analysis and experiment.

EFFECT OF SURFACE CHARACTER ON BOILING OF LIQUID METALS

A few years ago our laboratory investigated the influence of surface finish and surface activation agents on boiling heat transfer for various non-liquid metals. Figure 1 shows the nature of the results for varying surface finishes. The location of the nucleate and transition boiling curves is greatly influenced by surface finish, while the maximum heat flux in nucleate boiling, the minimum heat flux in film boiling and the position of the film boiling curves are relatively uninfluenced by surface finish.

Much recent data for boiling of liquid metals is difficult to understand. The location of the q/A *vs.* ΔT curves does not seem to be consistent in various apparatus. We propose to investigate for liquid metals (as we did for non-metals) the influence of surface finish and some additives on the location of the boiling curves.

FILM BOILING IN FORCED CONVECTION INSIDE TUBES

For the past year and a half we have been investigating film boiling inside vertical and horizontal tubes. The test section is 304 Stainless steel (0.402″ *ID*, 15″ long heated section) in which Freon-113 flows. In addition there is a parallel visual observation test section consisting of a glass tube (0.418″ *ID*, 4 ft long with a 9″ long heated section). The tube has a semi-transparent electrically conducting coating on the outside. The flow goes through either the test section or the visual observation section where the same thermal conditions are set.

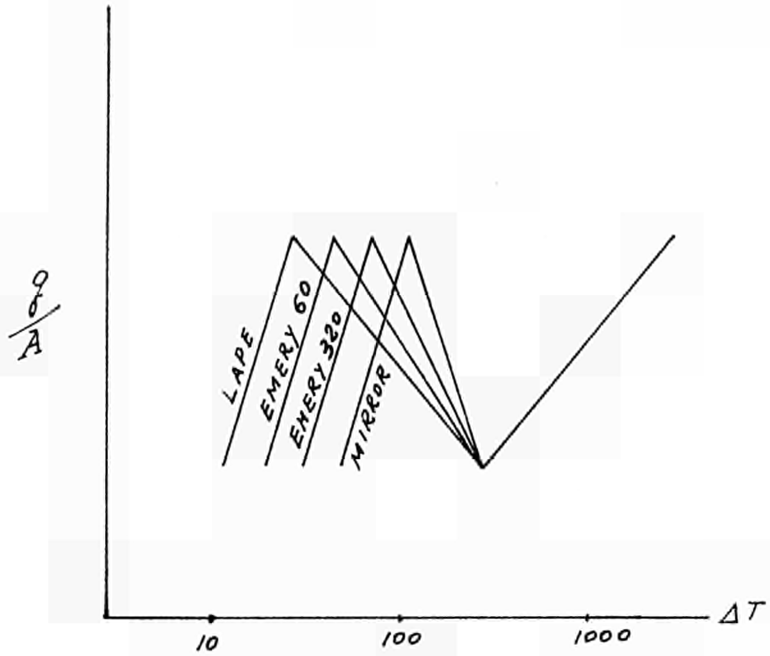


Figure 1

In the horizontal tube the flow was stratified for the low quality at exit in these tests. Figure 2 shows a sketch of the nature of the flow. Because of the non-symmetry the use of a heat transfer coefficient is not advisable. An analysis was developed for predicting the wall temperature distribution. In the lower portion of the tube the Bromley analysis for constant wall temperature on the outside of a cylinder was modified for the present case of uniform (q/A) inside. In the upper portion the heat transfer rate was predicted by use of an equivalent diameter for a circular arc segment and the ordinary forced convection relation for vapor flow. The flow areas for vapor and for liquid were determined by writing momentum equations in the liquid and in the vapor and equating the pressure drop in the two streams. Agreement between predicted and measured wall temperatures at the top and bottom of the tube was reasonably good.

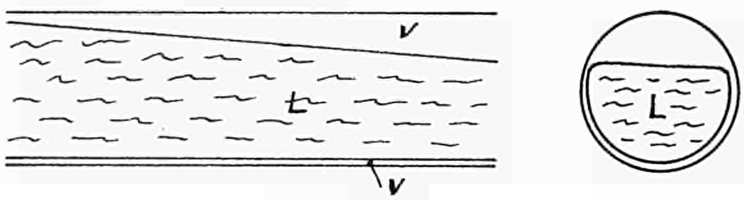


Figure 2

Currently we are studying the film boiling phenomena in vertical tube flow. Here there is a more or less symmetrically placed liquid core with a very rough surface. Here again we are attempting to predict wall temperature by writing momentum equations for the two streams accounting for gravity forces as well. The vapor in the annulus flows at much higher velocities than the liquid. Our current difficulty is in describing the flow resistance at the rough time-varying liquid-vapor interface.

Reports on this subject should be available in a few months.

EFFECT OF ELECTROSTATIC FIELD ON BOILING

During the past two years we have been investigating the various effects of electrostatic fields on boiling. One such investigation completed recently by Dr. H. Choi studied boiling of Freon-113 on a horizontal wire. Around the wire, co-axially, was a 1.5″ diameter electrically-conducting Pyrex tube. A radial electrostatic field was established between the wire and the co-axial cylinder.

The force on a dipole in a spatially varying electrostatic field is readily computed. This leads to the determination of the electrostatic force per unit volume or per unit mass of the fluid. This force acts like a radial "gravity" force and we compute the magnitude of g_e, the electrostatic "gee" at the surface for any field strength.

The results of the experiments are shown in figure 3. In the natural convection non-boiling regions to the left, the data were correlated by Grashoff and Prandtl numbers, as in gravity natural convection, but here g_e replaces the gravity g in the Grashoff number.

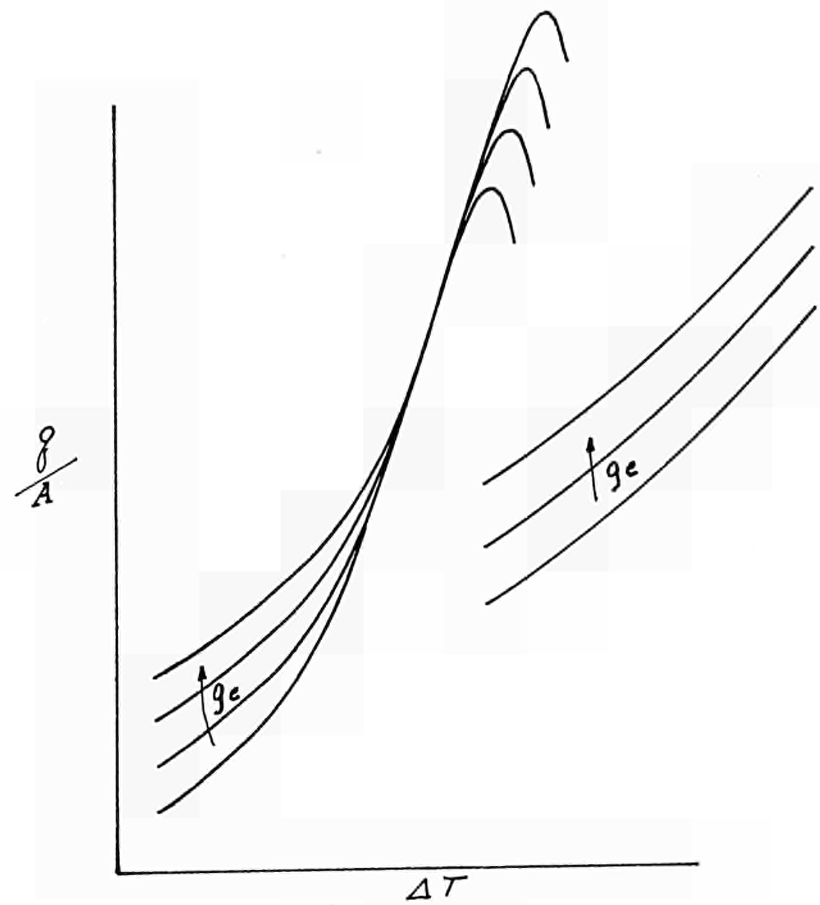


Figure 3

In the nucleate boiling zone the data merge and even cross over slightly as do the results of Merte and Clark for boiling in centrifugal fields.

The peak heat flux is found to agree reasonably well with the Zuber correlation and to increase with $g_e^{1/4}$ as expected from centrifugal field data.

In the film boiling regime the data are correlated reasonably well with the Bromley equation with g_e replacing g.

Work in this field continues under the direction of Dr. H. Choi at Tufts University, Medford, Massachusetts, and Dr. J.C. Reynolds in our laboratory.

SLUG CAPILLARY FLOW

For the past year Professor Griffith has been working on the problem of slug flow where the gas zones have well-defined boundaries as shown in figure 4. When $\frac{V^2 D \rho_l}{\sigma_{lv}} < 100$ the bubbles are separate and distinct. The liquid may be in a laminar or a turbulent flow but there is no vapor in the liquid, or slug, zone. When $\frac{V^2 D \rho_l}{\sigma_{lv}} > 100$ there is a bubbly wake behind the bubbles.

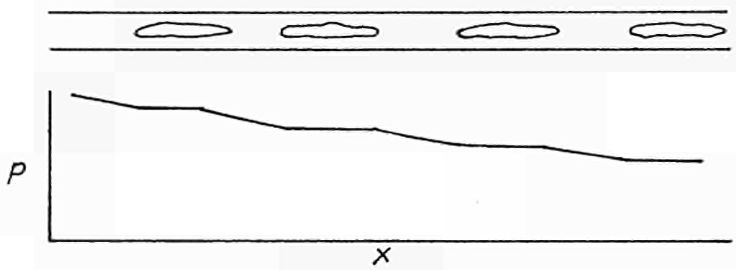


Figure 4

In the absence of a bubbly wake, the pressure distribution is as shown in figure 4 —uniform pressure through the bubble and a linear drop in pressure in the slug. At the lower velocities the pressure drop increases linearly with slug length. The bubbles move at a velocity of approximately 1.2 times the average liquid velocity.

This work continues and reports should be available by July 1963.

SLUG FLOW IN NON-CIRCULAR PASSAGES

Previous work has led to the following equation expressing the absolute bubble rise velocity:

$$V_{\substack{\text{bub.}\\\text{abs}}} = C_1 \sqrt{gD} + C_2 \frac{Q_l + Q_g}{A_p}$$

For circular tubes $C_1 = 0.35$ and C_2 is a weak function of Reynolds number but is approximately 1.2 over a wide range.

During the past year Professor Griffith has investigated the same phenomena in the following shaped passages (figure 5):

a) Rectangular

b) Annuli

c) Tube Bundles

In each case the result was expressed in terms of the above equation. C_2 was a weak function of Reynolds number but was approximated as 1.2. The coefficient C_1 was plotted as a function of D_1/D_2 and the D in the equation was D_2 shown in the sketches.

A report should be available in a few months.

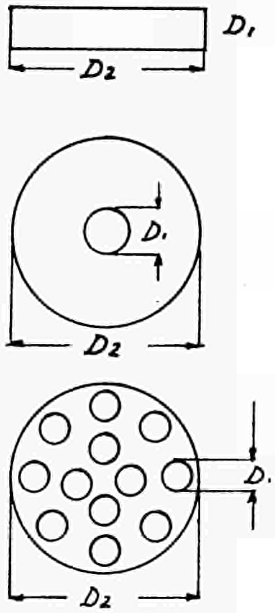


Figure 5

OTHER WORK RECENTLY REPORTED

Other work which has already appeared in reports but which should be of interest to this conference is as follows:

a) Report No. 8767-21 by A.E. Bergles contains criteria for determining the surface superheat required for nucleation under various flow conditions. Also there are further experimental results associated with the superposition concept in forced convection boiling. Another significant feature of this report is the sub-cooled forced convection burnout data for varying tube diameter from 0.180″ down to 0.027″. These data show $(q/A)_{max}$ to increase significantly as D decreases at constant G and L/D.

b) Report No. 7673-19 by C.Y. Han contains additional evidence justifying a criterion for nucleation. It also postulates a description of the mechanism of heat transfer associated with nucleate boiling in the lower flux region and justifies calculated results by experimental measures. The report also contains an analysis of bubble growth and departure for a bubble growing at a heated surface.

A STUDY OF CONVECTION BOILING INSIDE CHANNELS

J.K. FERRELL

North Carolina State College, Raleigh, North Carolina, USA

The purpose of this project is to carry out experimental and theoretical studies on two-phase, single component, frictional and expansion-contraction pressure losses. The program is essentially a fluid dynamics study, however, both heated and unheated channels are being investigated.

All experimental equipment is designed so that either water or Freon may be used as a fluid. Since the object is not to produce reactor design data, but to further the understanding of two-phase fluid dynamics, there are many advantages to using Freon as a working fluid. The two most important are the low-latent heat of vaporization and the fact that the full range of pressures from atmospheric to the critical pressure can be covered in relatively simple experimental equipment.

The experimental approach to the problem is the simultaneous measurement of all important parameters. Since the research is relatively low-pressure, visual observation at the test section exit is possible, and will provide for qualitative determination of the flow regime.

The experimental equipment consists of a stainless steel, closed loop, designed for pressures of up to 600 psia. The test section is stainless steel tubing 8′ long, $^1/_2''$ outside diameter with 0.03″ wall thickness. Expansion and contraction test sections will be unheated and will be installed at the exit of the heated section. The test section is heated with alternating current and with a power input of up to 150 K.W. Most of the loop control and reading of data is mannual.

The following parameters are measured or controlled by the method indicated:

1. Loop flow rate is controlled and measured with a turbine flow meter or with an orifice meter.

2. Pressure losses in the test section are measured with differential pressure cells or with manometers.

3. The steam quality is calculated from a heat balance.

4. Temperatures are measured with thermocouples.

5. Power input to the test section is controlled and is measured with standard electrical meters.

6. The steam volume fraction is measured by the gamma attenuation technique using Th-170 as a source and a scintillator as the detector.

At the present time, construction of the experimental equipment has been completed and some preliminary data have been obtained. These preliminary runs have indicated that the equipment is in working order and that the most difficult of the measurements, the steam volume fracion in the $^1/_2''$ diameter test section, can be accomplished with an accuracy of about $\pm$ 10%.

The next phase of the experimental program is the measurement of pressure losses for the various test sections over the full range of parameters possible with the equipment.

TWO-PHASE FLOW AND BOILING STUDIES

G.B. WALLIS

Dartmouth College, Hanover, New Hampshire, USA

INTRODUCTION

At Dartmouth College we are interested in fundamental studies of two-phase flow and boiling. By this I mean that we seek to express the values of particular dependent variables [e.g. pressure drop, burnout, stability, flow pattern, voidage] as functions of three types of basic parameters.

(1) The fluid properties (e.g. densities, viscosities, surface tension, conductivity, latent heat).

(2) The apparatus geometry (e.g. length, diameter).

(3) The independently controlled variables (e.g. inlet temperature, heat flux, flow rates).

We are particularly interested in problems for which there is a reasonable hope of obtaining an analytical solution and proving its validity in practice.

Our general philosophy of approach can perhaps best be demonstrated by referring to a specific problem —the hydrodynamics of two-phase flow in a vertical pipe—and describing what we already know about it and the directions in which we hope to make some further progress.

Figure 1 shows the typical series of flow patterns which occur in a vertical pipe in which boiling is taking place. Single phase liquid enters at the bottom and sooner or later bubbles begin to nucleate at the heated surface. These bubbles spread into the stream to form a *bubbly* flow pattern. Further up the tube the bubbles grow and coalesce until eventually large *slug flow* bubbles which fill the tube are formed. With continuing evaporation these slug flow bubbles become longer, occupy more of the pipe and eventually join to form

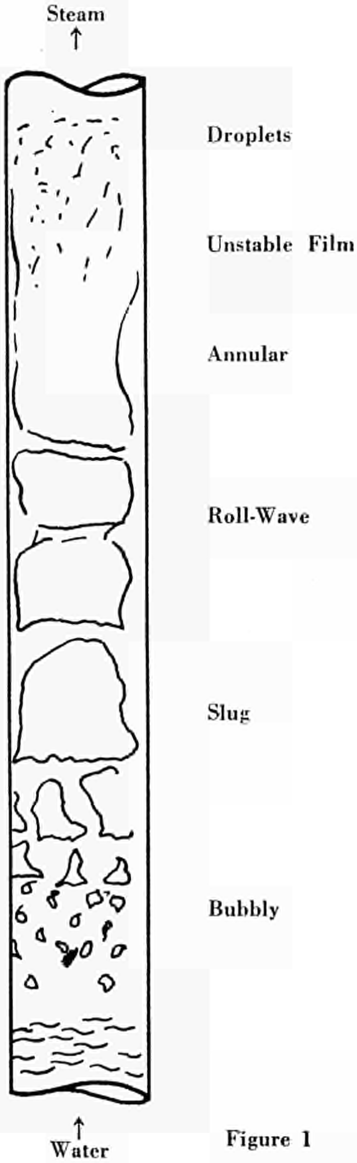


Figure 1

a continuous vapor core. However, the liquid flow is still downwards in the film and therefore water can only flow upwards in large "roll-waves" which could be regarded as degenerate liquid "plugs" which have been penetrated by the vapor core. At sufficiently high vapor velocities and low liquid rates the roll-waves disappear and a relatively wave-free smooth *annular film* is obtained. Further evaporation and increasing vapor velocity lead to an instability of the liquid film and the stripping of liquid filaments from its surface. Thus the vapor core becomes laden with *entrained droplets.* Eventually, when all the annular film has been evaporated, only the entrained droplets remain and these evaporate both by deposition on the wall and by conduction and radiation into the core. Eventually superheated steam is formed.

The above is only a simplified survey of what actually happens in practice. Under some circumstances flow regimes overlap or disappear completely while in other cases new patterns (such as "churn" flow) can occur. It is likely that different analytical expressions will describe the dependent variables in each flow regime.

SUMMARY OF PRESENT KNOWLEDGE AND SUGGESTIONS FOR FUTURE WORK

1 — SINGLE PHASE LIQUID FLOW is already fairly well understood and reliable equations exist for design purposes.

2 — BUBBLY FLOW

2.1 — *Present Knowledge*

By using the principle of relativity it has been possible to gain a reasonably good understanding of bubbly flow for the case in which wall shear stresses and variations in velocity and concentration across the channel are insignificant. If the "superficial" velocities (flow divided by total pipe area) of the vapor and liquid are V_g and V_f and the void fraction is α, then the relative or slip velocity is

$$V_r = \frac{V_g}{\alpha} - \frac{V_f}{1-\alpha} \tag{1}$$

It is sometimes convenient to clear of fractions to obtain an equation which defines a new "characteristic velocity", V_{fg},

$$V_{fg} = V_r\,\alpha(1-\alpha) = V_g(1-\alpha) - V_f\alpha \tag{2}$$

Then, by expressing V_{fg} as a function of the fluid properties, bubble size and voidage, it is possible to calculate "α" for any given values of V_f and V_g (references 1, 2, 3). Since wall shear stress has been neglected throughout the analysis the pressure gradient is

$$\frac{\Delta P}{\Delta L} = g[\rho_f(1-\alpha) + \rho_g\alpha] \tag{3}$$

It is readily shown (3, 4) that a "continuity shock-wave" between voidages α_1 and α_2 moves with a velocity

$$U_s = (V_f + V_g) + \frac{(V_{fg})\ \alpha = \alpha_1 - (V_{fg})\ \alpha = \alpha_2}{(\alpha_1 - \alpha_2)} \tag{4}$$

and that in the limit $\alpha_1 \rightarrow \alpha_2$ the velocity of a "continuity wave" is

$$U_w = (V_f + V_g) + \frac{dV_{fg}}{d\alpha} \tag{5}$$

Therefore the unsteady-state behaviour of such a system can readily be calculated.

It is empirically found that the characteristic velocity, V_{fg}, can be expressed as

$$V_{fg} = U_{b\infty}\ \alpha(1-\alpha)^{n+1} \tag{6}$$

where $U_b{}^\infty$ is the velocity of rise of a single bubble in an infinite stagnant medium and the index "n" is a function of Reynolds Number (3.5) [and to some extent also the Weber Number].

Under certain circumstances (1, 3) no value of α will satisfy both equations (2) and (6) and in this case bubbly flow cannot exist and another flow pattern occurs (usually slug flow).

Note that the bubble size, which it is necessary to specify before equation 6 can be completely solved, is a result of the inlet conditions and the past history of the flow [for instance the number of coalescences] and therefore the motion is not wholly determinate solely in terms of the flow rates V_f and V_g.

The general theory of one-dimensional vertical two-component flow [described by equations (1) and (6)] has also been useful for deriving an expression for the burnout heat flux in pool boiling (2) and for developing a simplified theory of sedimentation (3).

2.2 — *Opportunities for Further Work*

In order to improve our understanding of the bubbly flow regime it will be necessary to take account of two things :

(1) The effect of wall shear stress and shear distribution across the flow cross-section.

(2) The effect of velocity and density variations and the rate of bubble migration across the section under non-equilibrium [entrance region] conditions.

2.3 — *Transition from Bubbly to Slug Flow*

Because a large bubble has less surface area than the equivalent volume of small bubbles there is a tendency for bubbles to agglommerate. The bubbly regime is therefore usually metastable under all conditions, and eventually turns into slug flow in a very long pipe even when the volumetric flow rates are constant (1).

We therefore have to distinguish between the *possibility* of bubbly flow and its *stability* against coalescence. Studies of the development of slug flow from bubbly flow have been performed by Moissis and Radovcich [6]. The transition is sensitive to the liquid purity, the inlet conditions and the flow history.

3 — SLUG FLOW

3.1 — *Simplified Theory*

The fully developed slug flow regime was studied by Griffith and Wallis [7] who related the overall characteristics [voidage and pressure drop] to the motion of individual bubbles. The velocity of a cylindrical bubble relative to the liquid ahead of it was taken from the work of Dumitrescu [8]

$$U_b = 0.35\ \sqrt{gD} \tag{7}$$

If it is assumed that the velocity of a bubble is unaffected by the proximity of other bubbles [i.e. the liquid velocity profile in the slug is fairly flat] the value of V_{fg} can be determined from equation (2) by considering the particular case when a string of bubbles is brought to rest by a vertical downflow of liquid. The velocity of the liquid is then from (7)

$$V_f = -0.35\ \sqrt{gD} \tag{8}$$

and the bubble velocity is zero independent of voidage, therefore

$$V_g = 0 \tag{9}$$

Substitution in (2) from (8) and (9) gives

$$V_{fg} = 0.35\ \alpha\ \sqrt{gD} \tag{10}$$

The general theory may now be developed as in the case of bubbly flow for all values of α.

3.2 — *The Effect of Velocity Profile*

If equation (10) is substituted in (2) some rearrangement leads to :

$$\frac{V_g}{\alpha} = (V_f + V_g) + 0.35\ \sqrt{gD} \tag{11}$$

The left hand side of equation (11) represents the bubble velocity, U_b, and the right hand side shows that this is equal to the sum of the total average velocity [the water velocity in the liquid slug] and the bubble velocity relative to the liquid slug. This suggests that the effects of velocity profile on the bubble velocity could be rationalised by a correction factor applied to either the liquid velocity or the relative bubble velocity. Griffith and Wallis originally chose to modify the factor 0.35 in equation (10), thus

$$V_{fg} = C_2\ 0.35\ \sqrt{gD} \tag{12}$$

where C_2 was dependent on the Reynolds Number for the liquid in the slug. This method of correlation gave consistent results for laminar flow but was not conclusively successful for Reynolds Numbers greater than about 3000. A more successful correlation for turbulent flow was derived by Nicklin and Davidson [9] who chose to apply a correction factor to the liquid velocity. The modified form of equation (11) was

$$U_b = \frac{V_g}{\alpha} = 1.2\ (V_f + V_g) + 0.35\ \sqrt{gD} \tag{13}$$

The correction factor 1.2 may be related to the ratio between the maximum velocity to the average velocity for turbulent flow. The core of the liquid is moving at about 1.2 times the average velocity.

3.3 — *The Interaction between Bubbles*

The wake behind one bubble modifies the velocity profile which is "seen" by the following bubble. As a result the second bubble rises faster and tends to catch up with the first (7). This effect was rationalised by Moissis [10] in terms of the factor C_2 of equation (12), which was expressed in terms of the ratio between the bubble separation and the pipe diameter

$$C_2 = 1 + 8 \exp\left(-1.06\, Ls_{/D}\right) \tag{14}$$

(L_s = slug length or distance between bubbles)

In a developing slug flow pattern the mean bubble velocity is increased as a result of the wake effects which may persist for long distances (another example of entrance effects in two-phase flow).

3.4 — *The Effects of Surface Tension and Viscosity*

Equation (7) is only valid when viscosity and surface tension effects can be ignored. In the general case the rise of a cylindrical bubble in stagnant liquid in a tube is governed by the balance between four different forces, buoyancy, liquid inertia, surface tension and viscosity. The balance between buoyancy and the three other forces may be expressed as three dimensionless parameters

$$\frac{\rho_f U^2}{gD\,(\rho_f - \rho_g)}\,, \qquad \frac{\sigma}{gD^2\,(\rho_f - \rho_g)}\,, \qquad \frac{U}{gD^2\,(\rho_f - \rho_g)}$$

and the bubble motion will, in general, be a function of these quantities (11).

When inertia dominates the value of the relevant parameter is

$$U\,\rho_f^{1/2}\,[gD(\rho_f - \rho_g)]^{-1/2} = 0.35 \tag{15}$$

which is consistant with equation (7) for low gas densities.

When surface tension dominates the bubble does not move at all, this occurs when

$$\frac{gD^2\,(\rho_f - \rho_g)}{\sigma} < 3.37 \tag{16}$$

When viscosity dominates the rise velocity is given by

$$U/D^2 g\,(\rho_f - \rho_g) = 0.010 \tag{17}$$

General correlations which account for the effects of all three forces acting simultaneously are given in reference (11).

3.5 — *Suggestions for Future Work*

1. Equations (7) and (11) or (13) can be adapted to tubes of different geometry such as square or rectangular sections or tube bundles. Some work of this kind is being undertaken by Griffith at MIT.

2. The effects of surface tension and viscosity are at present only understood for stagnant liquid. Their influence on voidage and pressure drop in flowing systems needs to be investigated.

3. Better predictions of the conditions under which slug flow occurs are necessary so that the designer knows which equations to apply. The prediction of flow regime is, of course, a general problem in two-phase flow work.

4 — THE TRANSITION FROM SLUG TO ANNULAR FLOW

4.1 — *The Upper Limit of Slug Flow*

It was suggested by Nicklin and Davidson [12] that the upper limit of slug flow was due to instability of the liquid film running down the outside of a bubble. It is observed in practice that when a film of liquid flows down the inner wall of a tube while gas flows upwards in the centre there are clearly-defined limits inside which the flow is stable. The onset of instability is known as "flooding". The stability limits are shown in Figure 2.

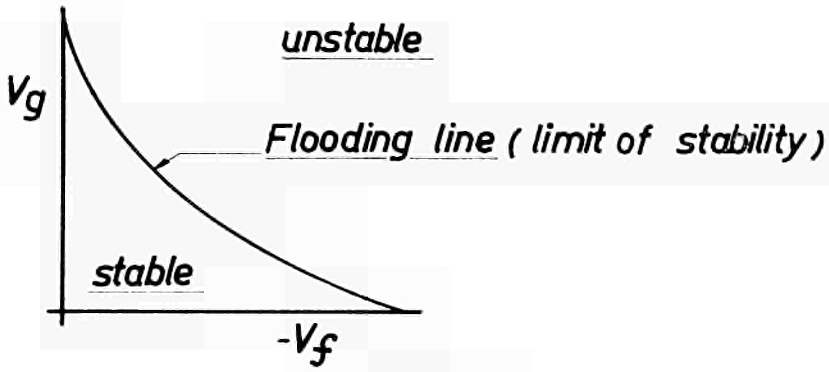


Figure 2

Dimensionless correlation of the flooding line (stability limit) has been attempted using the parameters

$$V_g^* = V_g \rho_g^{1/2} [gD(\rho_f - \rho_g)]^{-1/2} \tag{18}$$

$$V_f^* = -V_f \rho_f^{1/2} [gD(\rho_f - \rho_g)]^{-1/2} \tag{19}$$

and the results appeared to follow the curves

$$V_f^{*1/2} + V_g^{*1/2} = C \tag{20}$$

where C was a constant depending on the method of introduction and removal of the gas and liquid (13).

When care was taken to eliminate all end-effects and extraneous causes of instability a value of $C = 1$ was obtained (14). It was noticed that once flooding has occurred and a "churn" flow pattern had been formed the velocity of one component or the other had to be reduced below approximately the line given by $C = 0.88$. This "hysteresis" demonstrated that, over a certain range, the particular pattern obtained in practice depended on the way in which the operating point was approached.

If a correlation of the form of equation (21) can be shown to be universally valid then presumably the upper limit of slug flow can be predicted. Equation (21) is so simple that there appears to be reasonable cause for hope that it will eventually be explained theoretically in terms of the wave motion of the interface.

4.2 — *Roll-wave Flow*

When the slug flow bubbles join to form a continuous gas core the liquid slugs degenerate into large roll-waves riding on the liquid film. Flow in the smooth parts of the film is downwards whereas the liquid waves move upwards. The net flow is a combination of these processes. In order to understand this regime we need to investigate the laws of motion of the individual waves (15).

4.3 — *The Onset of Annular Flow*

True annular flow can be said to occur when there is no longer any downwards motion in the liquid film. It has been suggested (16) that this point can be defined in terms of the limit of the flooding line at zero liquid velocity. Equations (18) and (20) then lead to

$$V_{g\,(\text{onset of annular flow})} = C^2\,[gD\,(\rho_f-\rho_g)]^{1/2}\rho_g^{-1/2} \qquad (21)$$

with a value of C^2 between 0.8 and 1.0. Further work is necessary to show the limitations of this formula.

5 — ANNULAR FLOW

Annular flow has been studied more than any other regime. However, our understanding is limited at present because of our inability to take account of the wave motion of the interface which may be the dominant phenomenon in some cases [17]. It again appears that an understanding of the types of wave motion which occur at fluid interfaces is the key to the development of a successful theory.

6 — THE ONSET OF DROPLET ENTRAINMENT

A new kind of instability occurs at the highest vapor velocities when the liquid is stripped from the film surface by the high shear stress and droplets are entrained in the main stream. This phenomenon is not understood but recent experiments (18) indicate that entrainment starts at a clearly defined gas velocity which is relatively insensitive to liquid velocity, pipe orientation or length, inlet conditions, or pipe size. Since droplets presumably originate from protuberances on the liquid surface we are again led to a study of the wave motion and stability of the interface. One of our main efforts at Dartmouth will be concentrated on this problem since the rewards for its solution are considerable [it may also explain a mechanism for burnout].

REFERENCES

[1] WALLIS, G.B. — *Int. Heat Transfer Conference,* Boulder, Colorado. Paper No. 28, 1961.

[2] WALLIS, G.B. — "Two-Phase Flow Aspects of Pool Boiling from a Horizontal Surface", Paper No. 3, *Two-Phase Flow Symposium,* Inst. Mech. Engrs. London. February 1962.

[3] WALLIS, G.B. — "A Simplified One-Dimensional Representation of Two-Component Vertical Flow and its Application to Batch Sedimentation" 3rd Congress, *European Federation of Chemical Engineering, London.* June 1962.

[4] WALLIS, G.B. — "One-Dimensional Waves in Two-Component Flow", *AEEW-R162,* Winfrith Heath, England. April 1962.

[5] ZUBER, N. and HENCH, J. — "Steady State and Transient Void Fraction of Bubbling Systems and their Operating Limits", *General Electric Co.,* Report No. 62 GL 100, July 1962.

[6] RADOVCICH, N.A. and MOISSIS, R. — "The Transition from Two Phase Bubble Flow to Slug Flow", *M.I.T.*, Report No. 7-7673-22, June 1962.

[7] GRIFFITH, P. and WALLIS, G.B. — "Two-Phase Slug Flow", *Trans. ASME,* Series C, **83**, No. 3, p. 307, 1961.

[8] DUMITRESCU, D.T. — "Stromung an einer Luftblase in senkrechten Rohr", *ZAMM* **23**, p. 139, 1943.

[9] NICKLIN, D.J., WILKES, J.O. and DAVIDSON, J.F. — "Two-Phase Flow in Vertical Tubes", *Trans. Inst. Chem. Engrs.* **40**, p. 61, 1962.

[10] MOISSIS, R. and GRIFFITH, P. — "Entrance Effects in a Two-Phase Slug Flow", Paper No. 61-SA-30, *ASME,* 1961.

[11] WALLIS, G.B. — "General Correlations for the Rise Velocity of Cylindrical Bubbles in Vertical Tubes", *General Electric Co.,* Report No. 62 GL 130, August 1962.

[12] NICKLIN, D.J. and DAVIDSON, J.F. — "The Onset of Instability in Two-Phase Slug Flow", *Two-Phase Flow Symposium,* Inst. Mech. Engrs. London, February 1962.

[13] WALLIS, G.B. — "Flooding Velocities for Air and Water in Vertical Tubes", *AEEW-R123,* Winfrith Heath, England. December 1961.

[14] WALLIS, G.B. and HEWITT, G.F. — "A Study of Flooding during Countercurrent Gas-Liquid Flow in a Vertical Pipe Using Porous Sections for Introducing and Extracting the Liquid", *AERE-R4022,* Harwell, England (in preparation).

[15] WALLIS, G.B. — "The Influence of Liquid Viscosity on Flooding in a Vertical Tube", *General Electric Co.,* Report No. 62 GL132, August 1962.

[16] WALLIS, G.B. — "The Transition from Flooding to Upwards Concurrent Annular Flow in a Vertical Pipe", *AEEW-R142,* Winfrith Heath, 1962.

[17] LACEY, P.M.C., COLLIER, J.G. and HEWITT, G.F. — "Climbing Film Flow", *Two-Phase Flow Symposium,* Inst. Mech. Engrs. London. February 1962.

[18] WALLIS, G.B. — "The Onset of Droplet Entrainment in Annular Gas-Liquid Flow", *General Electric Co.,* Report No. 62 GL127, August 1962.

II. EURATOM TWO-PHASE FLOW PROGRAM

RESEARCH PROGRAM ON HEAT TRANSFER AND STABILITY PROBLEMS IN BOILING WATER REACTORS

M. BOGAARDT and C.L. SPIGT
Technological University, Eindhoven, The Netherlands

1. INTRODUCTION

In the laboratory of Heat Transfer and Reactor Engineering of the Technological University of Eindhoven a research program is under way to study the steady state and dynamical behaviour of a boiling water reactor. Experiments are being carried out on a pressurized water loop to study the hydraulics of a unit-cell of a boiling water reactor. Furtheron a combination will be made of the loop with an analogue computer simulating the neutronic feed back in order to study the behaviour of the total reactor system. It is the purpose to derive a physical description of the system, starting from the fundamental laws of conservation, explaining the results of the experiments. To this end use will be made of an analogue and digital computer available at the University.

Besides the experimental side, basic theoretical studies are being carried out for deriving the characteristic quantities describing a two-phase flow system. These studies will be backed up by some fundamental experiments using air-water systems as well as atmospheric boiling loops.

In the following the loop, test section and measuring methods will be specified in more detail, while also a layout will be given of the program and some results obtained so far will be mentioned.

2. DESCRIPTION OF THE LOOP

A flow scheme of the natural circulation pressurized boiling loop is given in Figure 1.

The test section consists of an electrical heating element centrally placed in a shroud. The steam-water mixture flows by natural convection through this annular passage.

The downcomer is the annular passage between the shroud and the steel wall of the 40 atmospheres pressure vessel. The steam produced in the test section is separated

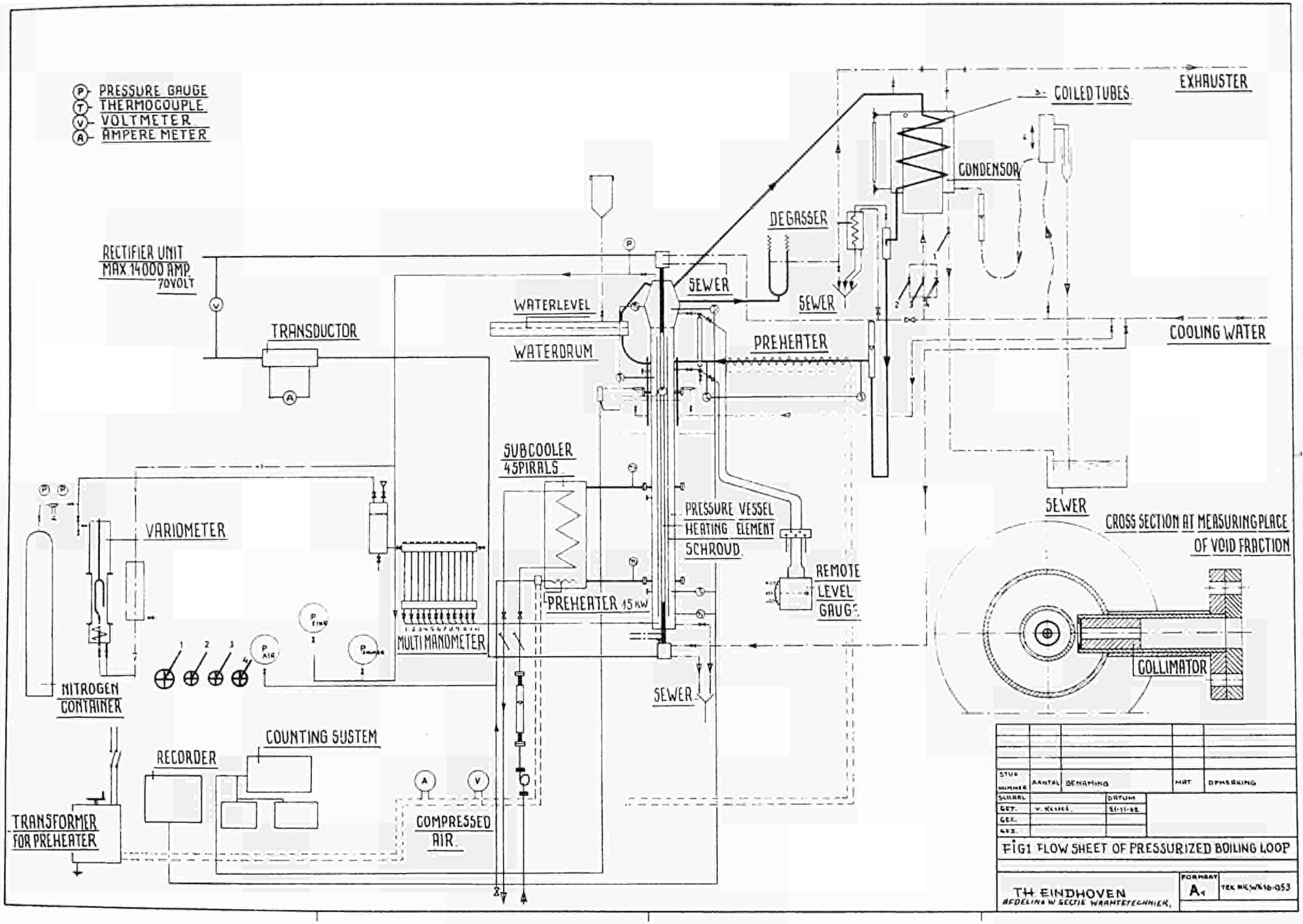


Figure 1

Flow sheet of pressurized boiling loop.

from the water flow at the top of this riser, flows to the condensor and the condensate is then returned to the downcomer through a preheater.

The pressure vessel is made of stainless steel. The cylindrical part has an inside diameter of 150 mm and is 3000 mm long. The water level is kept within certain limits by means of a water drum parallel to the test section. The steam is condensed inside three coiled tubes and makes evaporate part of the cooling water to the atmosphere. Condensor control can be achieved by automatic as well as by manual control.

The dummy fuel elements are heated by direct current fed from two rectifiers. An interesting feature of the power supply is the possibility of fast control.

Recently a subcooler circuit, see figure 1, has been added which gives the possibility to control the subcooling of the water at the inlet of the riser.

By incorporating a pump it has been made possible to use this circuit for forced circulation measurements. With this equipment it will be possible to separate the influencing parameters on the different fundamental quantities.

Two atmospheric glass loops have been constructed to study atmospheric boiling in more detail. These glass loops can be used either for water-steam or water-air mixtures.

The limits of the independent variables in the experiments will be

Power	0 - 1000 kW
Pressure	0 - 35 ata
Subcooling	0 - 350 kW
Mass flow	0 - 35 m^3/h (Forced circulation)

3. SPECIFICATION OF THE TEST SECTION

Most of the experiments will be done with a single rod stainless steel heating element with nominal diameter of 30 mm and a heated length of 2400 mm. Uniform as well as stepwise heat flux distributions simulating the cosine distribution in a reactor will be used. Some experiments will be carried out using a 7-rod bundle heated element with nominal diameter of the rods of 12.7 mm and a heated length of 1800 mm.

The shrouds are made of glass, aluminium or stainless steel and inner diameter of nominal 50, 60 and 70 mm. Some shrouds have open ends, others have the inlet and outlet holes in the hull.

4. QUANTITIES TO BE MEASURED

The following quantities are measured:

a) *Power*

The heat generation and the power of the preheater are measured electrically by means of precision instruments with mirror-reading. In non-steady states measurements are taken using Hall generators.

b) *Pressure*

The absolute pressure is measured using a Bourdon type gauge. The relative change with respect to the time is measured by a differentiating manometer. Static pressures along the riser and downcomer are made visible on a multimanometer. In dynamic experiments use will be made of capacitance pressure gauges developed in this laboratory.

c) *Temperatures*

All steam and water temperatures are measured by fast thermocouples. For obtaining information on the temperature of the heating element in steady and non-steady states the elongation of he heating element is measured and the signal from the burnout detector analyzed.

d) *Flow rate*

The circulation velocity is measured using static pressure taps or pitottubes in the lower end of the coolant channel. In steady state the multimanometer (see b) is used, while in non-steady states a capacitance differential pressure gauge is applied, that was developed in this laboratory. For forced circulation flow measurement a turbine flowmeter will be installed.

e) *Void fraction*

The void fraction along the channel is measured using the radio active and the capacitance method. These measuring methods are being applied both in steady and non-steady state. An acoustical method is under study as an alternative method.

f) *Condensate flow*

A normal magnetic type flow meter is used. The "subcooled power" is determined from the temperature difference across the riser and the flow rate in the channel.

For the purpose of recording and analyzing purposes the following equipment is or will be available:

Millivolt and temperature recorders.
Light-galvanometer recorder.
Magnetic tape recorder, frequency modulated.
Transfer function analyzer.
Noise correlator.

For the steady state *direct print-out* equipment has been ordered. For analysis of the results both analogue and digital computer are used.

5. PROGRAM

a) *Steady State*

In the steady state measurements the influence of pressure, subcooling, heat production, heat flux distribution and waterlevel on the dependent variables such as natural circulation flow rate, void fraction, slip factor, non-boiling length and two phase pressure drop are investigated. This is being done for different geometries, while also the influence of internal resistances placed in the channel will be studied.

At some combinations of heat production, subcooling and pressure instabilities occur. The starting point, the origin and the character of these instabilities are under study.

Forced circulation experiments will be carried for separating the influence of flow rate on the characteristic quantities.

It is conceivable that some studies on burnout will also be made.

b) *Non-steady state*

In the dynamic experiments the transfer functions from power, pressure and sub-cooling to the characteristic variables for heat transfer as flow rate, void fraction and temperatures are determined. Attention will also be paid to the transient response of the variables to step and other inputs. Noise analysis will be made using the crosscorrelation techniques.

In some experiments the neutronic characteristics of a reactor will be simulated by means of an analogue computer which controls power supply of the boiling water loop.

c) *Glass loop experiments*

Atmospheric glass loops have been constructed to study in more detail atmospheric boiling, including such phenomena as bubble formation, growth and detachment, separation of steam and water and boiling noise. These glass loops can be used either for water-steam or water-air mixture and are also used for calibrating purposes. For the experiments just mentioned a high speed film camera is available.

6. THEORETICAL STUDIES

The experiments carried out are backed up by theoretical studies. The purpose thereof is to try and to give an explanation of the observed phenomena and to derive a mathematical description of the loop behaviour. It is expected that this description will be subject to a continuous development, when more results are becoming available and more insight in the problem is obtained.

A digital computer program has been made for calculating the recirculation in loops. In this programme subcooled boiling is incorporated, while the heat flux distribution and correlations for slip factor and pressure losses can be fed in explicitely.

Besides this, the geometry and operating conditions can easily be varied.

A theoretical study has been started on the behaviour of a two-phase flow system. The equations for the non-steady state have been formulated from the basis of theoretical fluid dynamics.

Much effort has been spent on a good description of the increased frictional pressure drop and heat transfer under boiling conditions. Similarity parameters have been derived which govern the solutions of the exact equations. For the steady state case these similarity parameters are used for plotting the experimental results.

A survey is made of the existing literature on the description of the behaviour of a loop in transient conditions. Starting from the geometry and operating conditions of the existing loop a computer program is now being made for calculating the dynamic behaviour of the loop, by applying the principles of conservation. An analogue and a digital computer are used for this purpose. The results will be compared directly with the experiments.

7. RESULTS

Most of the work done so far was concentrated on the construction of the loop and the development of the measuring methods.

The results of a first series of experiments of void fraction and flow rate measurements as a function of power and pressure have been presented to the Two-Phase Flow Symposium of the Institute of Mechanical Engineers in Londen, February 1962, Reference 1.

Recently burnout and instability experiments have been carried out with a seven-rod bundle heating element, references 2 and 3.

An experimental program is now under way with a single rod heating element and a 50 mm shroud. Steady state and hydraulic instability experiments are being carried out.

For further information reference is made to the quarterly progress reports.

8. REFERENCES

8.1. *Quarterly Progress Reports*

Anonymous,

Quarterly Progress Report I. Research Program on heat transfer in a boiling water reactor, January 1st - April 1st 1961, Report WWO16 - R7, 1961.

Anonymous,

Quaterly Progress Report II. Research Program on heat transfer in a boiling water reactor, April 1st - July 1st 1961, Report WWO16 - R8.

Anonymous;

Quarterly Progress Reports III and IV. Heat transfer and stability studies in boiling water reactors, July 1st 1961 - January 1st 1962, Report WWO16 - R9.

Anonymous,

Quaterly Progress Report V. Research Program on heat transfer in a boiling water reactor, January 1st - April 1st 1962, Report WWO16 - R15, 1962.

Anonymous,

Quaterly Progress Report VI. Research Program on heat transfer in a boiling water reactor, April 1st - July 1st 1962, Report WWO16 - R18, 1962.

Anonymous,

Quaterly Progress Report VII. Research Program on heat transfer in a boiling water reactor, July 1st - October 1st 1962, Report WWO16 - R24.

Anonymous,

Quarterly Progress Report VIII. Heat Transfer and stability studies in boiling water reactors, October 1st - January 1st 1963, Report WWO16 - R30, 1963.

8.2. *Special technical reports*

Spigt, C.L., Simon Thomas, J.P., Bogaardt, M.,

Introductory laboratory studies of boiling reactor stability, Report WWO16 - R10, 1961. (Presented at the two flow symposium of the Institute of Mechanical Engineers, London, February 1962.) (Special technical report I.)

Van der Walle, F.

Study of possible application of accoustical methods for determining void fraction in boiling water reactors, Report WWO16 - R11, 1962. (Special technical report 2.)

Spigt, C.L.,

Results of burn-out and instability experiments on a 7-rod bundle at up to 30 atmospheres pressure and under conditions of natural convection and zero inlet subcooling, Report WWO16 - R21, 1963. (Special Technical report 3.)

Tummers, J.F., Spanjers, Th.H.

Results of flow rate measurements on a seven-rod bundle element with natural circulation and with inlet subcooling up to 30 atmospheres pressure, Report WWO16 - R26, 1963.

Spigt, C.L.

Instability and burn-out observations, made during a development of a burnout detector, Report WWO16 - R27, 1962.

Van der Walle, F.

On the design of an experimental acoustical set-up, Report WWO16 - R28, January 1963. (Special Technical Report 3.)

Spigt, C.L.

Results of burn-out and instability experiments on a 7-rod bundle at up to 30 atmospheres. Second series with inlet subcooling and natural convection, Report WWO16 - R31, 1963.

8.3. *Memoranda*

Hardy, M., Boullet, J.J., Everhard, H.H.

Prise de pression, memorandum WWO16 - M1, 1960.

Hardy, M., Simon Thomas, J.P., Spigt, C.L.

Refroidissement de la boucle thermique à circulation naturelle, memorandum WWO16 - M2, 1961.

Spigt, C.L.,

Rekenmodel I voor de bepaling van de natuurlijke circulatie in een kokend water systeem, memorandum WWO16 - M3, 1962.

Spigt, C.L.,

Rekenmodel II ter bepaling van de natuurlijke circulatie in een kokend water systeem, memorandum WW016 - M4, 1962.

Dijkman, F.J.M.,

Aanvulling memo's 3 en 4, betreffende slip-factor en 2-phase weerstand, memorandum WWO16 - M5, 1962.

Wien, T.

Short instruction manual for burn-out detector, memorandum WWO16-M6, 1962.

Bowring, R.W. Spigt, C.L.

Instability observations using a burn-out detector with a test-section of H.B.W.R. II, geometry in the Eindhoven loop, memorandum WWO16 - M7.

Everhard, H.H.

Rapport differentiaal drukopnemers, memorandum WWO16 - M9, 1962.

Tummers, J.F.

IJking van instroomverliesfactor in de kookproefopstelling, memorandum WWO16 - M11.

Tummers, J.F.

IJking van de instroomverliesfactor in de kookproefopstelling voor de Halden II B serie, memorandum WWO16 - M12, 1962.

Anonymus,

Summary of the results, obtained on December 31st, 1960 of the experiments to be carried out under contract with the Euratom U.S.A. Joint Board Research and Development Board, memorandum WWO16 - M13, 1961.

van Vlaardingen, H.F.

Scintillatie tellers, memorandum WWO16 - M14.

Spanjers, Th.H.

Berekening warmteverliezen van de kookproef met onderkoelingscircuit, memorandum WWO16 - M15.

Dijkman, F.J.M., Callen, J.D.

Rekenmodel III, Determination of the natural circulation velocity in a boiling water loop, memorandum WWO16 - M16.

Wamsteker, A.J.J., v. Vlaardingen, H.F.

Ontwikkeling van apparatuur t.b.v. dampfractie metingen met behulp van een radio-actieve methode, memorandum WWO16 - M17.

8.4. *Halden project reports and memoranda*

Bowring, R.W., Spigt, C.L.

Interim report on Halden II-7-rod bundle burn-out tests up to 30 atmospheres pressure. First series with zero inlet subcooling and natural convection, Report HIR-40, 1962.

Bowring, R.W., Spigt, C.L.

Halden II-7-rod bundle stability and burn-out tests up to 30 atmospheres pressure. Second series with natural convection and inlet subcooling, Report HIR- 1963.

Bowring, R.W., Spigt, C.L.

Instability observations using a burn-out detector with a test section of H.B.W.R. II geometry in the Eindhoven loop, memorandum HP-309, 1962.

Bowring, R.W.

Description of H.B.W.R. burn-out experiments to be carried out at Eindhoven, memorandum HP-279, 1962.

Bowring, R.W.

H.B.W.R. II Eindhoven burn-out experiments—instrumentated fuel assembly, memorandum HP-290.

Bowring, R.W.

H.B.W.R. II Eindhoven burn-out experiments—results of experiment I, memorandum HP-281, 1961.

Bowring, R.W.

Proposed second series of H.B.W.R. 7 rod bundle burn-out experiments, memorandum HP-319, 1962.

INSTABILITIES IN BOILING WATER LOOPS

S. FABREGA
Nuclear Research Center (CENG), Heat Transfer Section, Grenoble, France

INTRODUCTION

The experimental work described in the present report relates to a basic study of the phenomena known as *instabilities* in boiling water reactors, viz. the occurrence of dangerous oscillations in the power and flow rate beyond a certain power level.

It is now agreed that the phenomenon is purely hydrodynamic in origin and that the investigation may be carried out in out-of-pile loops heated at a constant power.

The experimental work carried out consists of three parts, viz:

a) a study of the pressure losses along a vertical boiling-liquid channel;

b) a study of the forced convection instability thresholds on an 8-atmosphere loop;

c) a study of the instability thresholds on a 1-atmosphere loop.

The following are the main features of the out-of-pile loops employed:

— they can be operated by means of natural or forced convection with a by-pass having a high flow rate and large section;

— simple and variable geometries (variable inclination on a 1-atmosphere loop);

— low pressure;

— very complete measuring equipment and sighting possible on the 1-atmosphere loop.

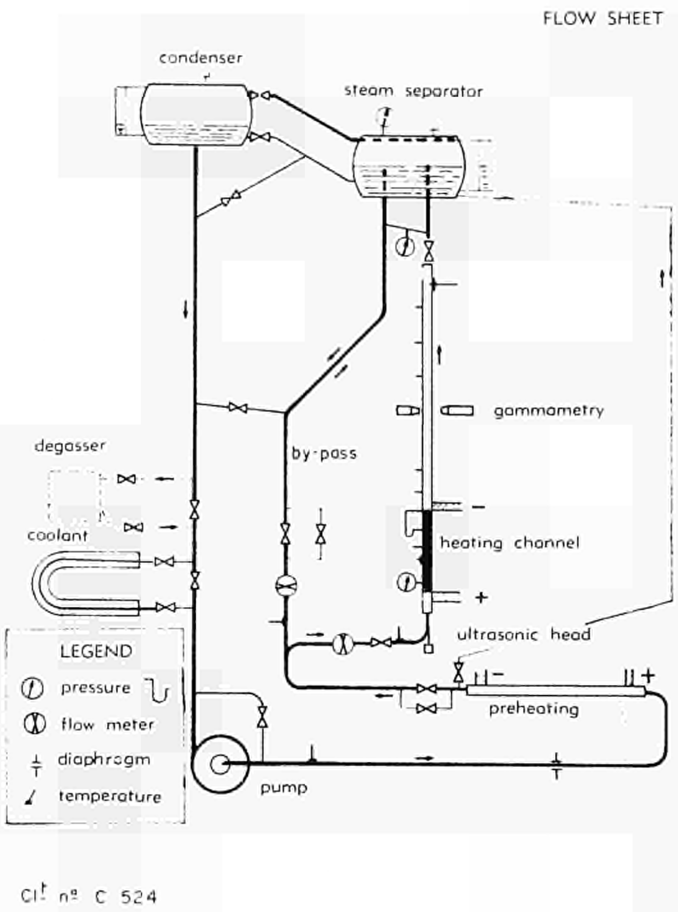
See Figure 1.

1. A STUDY OF THE PRESSURE LOSSES

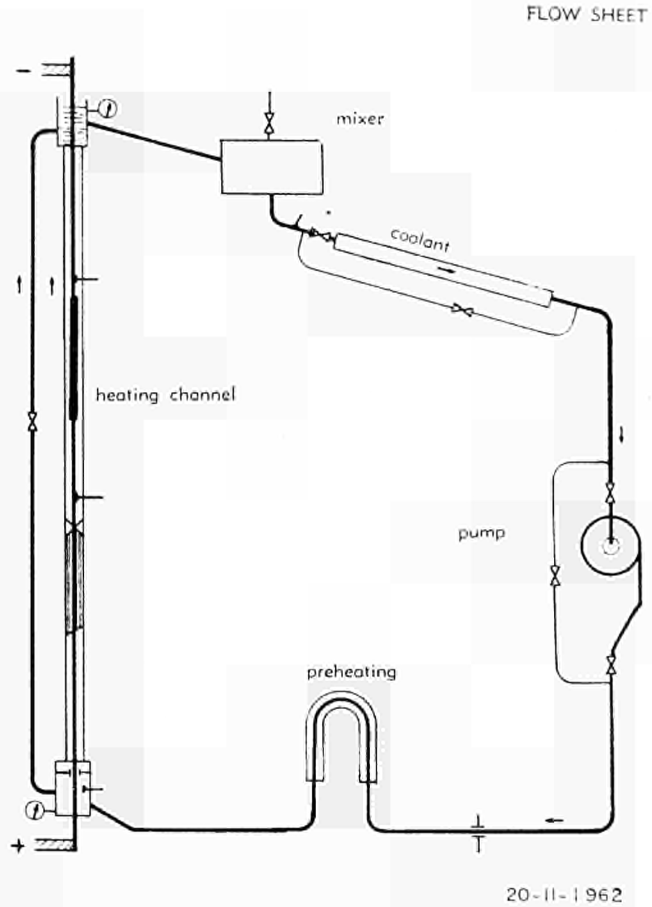
The study on the 8-atmosphere loop was undertaken with a view to employing the *effective* laws governing pressure drop in a mathematical model which is now being investigated to arrive at an explanation of the instabilities.

1.1. *The experimental data are:*

— a vertical, cylindrical channel with a circular cross-section 6 mm in diameter; upward flow; wall of A4 drawn commercial available aluminium (relative roughness about 8.10^{-4});



Boiling water loop CF 4



Visualisation loop

Figure 1

— surface density of thermal flux: from 5 to 30 W/cm^2;

— liquid velocity at inlet: 1 to 4 m/sec;

— absolute pressure at the inlet: 6 to 10 kgf/cm^2;

— quality at the outlet: 0.5 to 5%;

— bipermutited degassed water.

1.2. *Results*

The Martinelli-Nelson parameter $\varnothing_{l_0}$ was calculated as a function of the density of the emulsion with respect to the liquid $\left(\frac{\bar{\rho}}{\rho}\right)$ and as a function of the Martinelli-Nelson parameter X_{tt}.

It will be noted that:

a) the void function was calculated with allowance for local boiling and with the use of the Armand model,

b) the pressure drop due to friction was deduced from the impulse equation.

The chief findings were as follows:

a) in the plane $\log \varnothing_{l_0}^2 \ \log \frac{\bar{\rho}}{\rho}$ the slope of the mean correlation straight line depends on the method of calculating α.

b) wall heating has a very marked effect. In graph 2, which shows $\varnothing_{l_0}^2$ as a function of X_{tt} we have grouped the results as a function of the boiling parameter $Eb = \frac{\varphi(u' - u)}{\alpha V_E}$. There appears to be a considerable increase in $\varnothing_{l_0}^2$ with Eb.

2. A STUDY OF THE INSTABILITIES ON THE 8-ATMOSPHERE LOOP

2.1. *Principles underlying the investigation*

From the point of view of the conditions at the limits, there is an analogy between natural convection operation and forced convection operation with a by-pass having a high relative flow rate.

In both cases the conditions at the limits defining the operating conditions of the channel-riser assembly are:

a) the specific heating power;

b) the inlet pressure;

c) the degree of subcooling at the inlet;

d) the pressure drop imposed along the channel-riser assembly and independent of the flow rate at the channel inlet (in the case of natural convection this is the hydrostatic drop in the cold downcoming column);

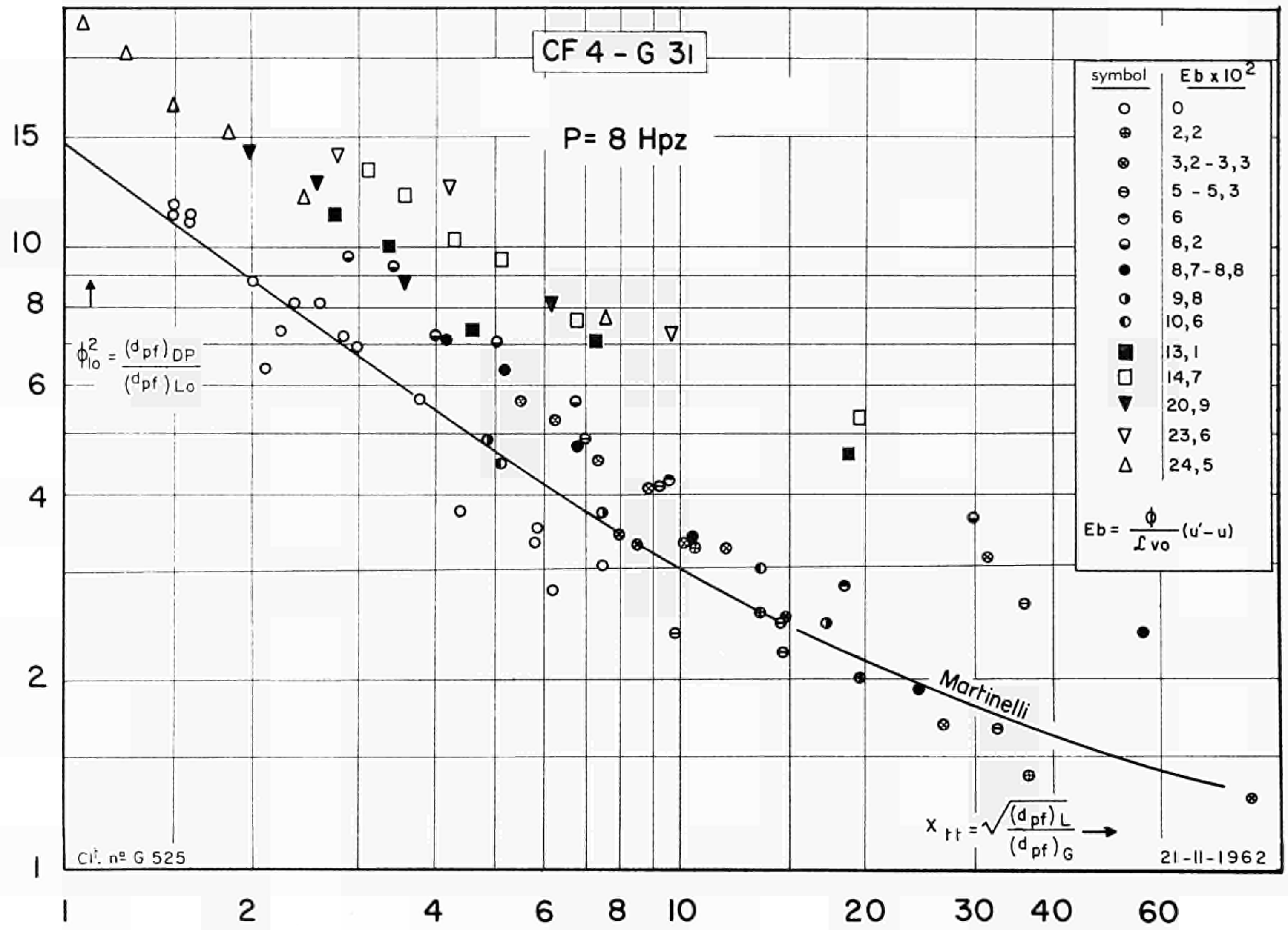


Figure 2
Friction during boiling

e) terms of pressure drop and hydraulic inertia, which depend on the flow rate in the channel and determine (when subtracted from the pressure drop imposed) the effective and instantaneous pressure drop along the channel-riser assembly.

These two terms are in fact geometrical parameters relating to the natural convection return circuit, and the connecting piping between the channel inlet and the by-pass in the case of forced convection. We have expressed them as channel lengths giving the same pressure drop or the same liquid phase hydraulic inertia for the same flow-rate values.

f) Finally, the geometry of the channel, particularly the hydraulic diameter and the length of the riser may also play a part.

All these parameters are independent of the flow rate oscillations in the channel.

The other parameters, e.g. steam quality at the outlet or the speed, were measured directly or calculated from the preceding conditions.

2.2. *Chief Results Obtained*

a) INSTABILITY THRESHOLDS

Graph 3 shows boundaries at which oscillations occur under given conditions of geometry and pressure.

Each curve corresponds to a value of the pressure drop involved.

The results are reproducible and the boundary is the same for any direction or manner of progression (by increasing or decreasing the power, temperature, pressure drop etc.). Moreover several types of oscillation phenomena were observed, viz:

— progressive or non-progressive occurrence;

— occurrence for a steam quality close to zero at the channel outlet and without previous redistribution of the flow rate in the channel;

— occurrence, after redistribution of the flow rate in the channel and for a steam quality of a few per cent at the channel outlet.

b) OSCILLATIONS PERIOD

This is in the range of 1 to 10 seconds. The period in the region of the threshold increases with decreasing power for a given value of the pressure drop imposed.

c) FORM OF THE OSCILLATIONS

Recordings 4 and 5 show the following general features:

— regularity of the phenomenon,

— marked dissymetry of the oscillations.

d) EFFECT OF A SINGLE PRESSURE DROP AT THE CHANNEL INLET

Other conditions remaining the same, it was found that there was an increase in the steam quality at the instability threshold and a lengthening of the periods.

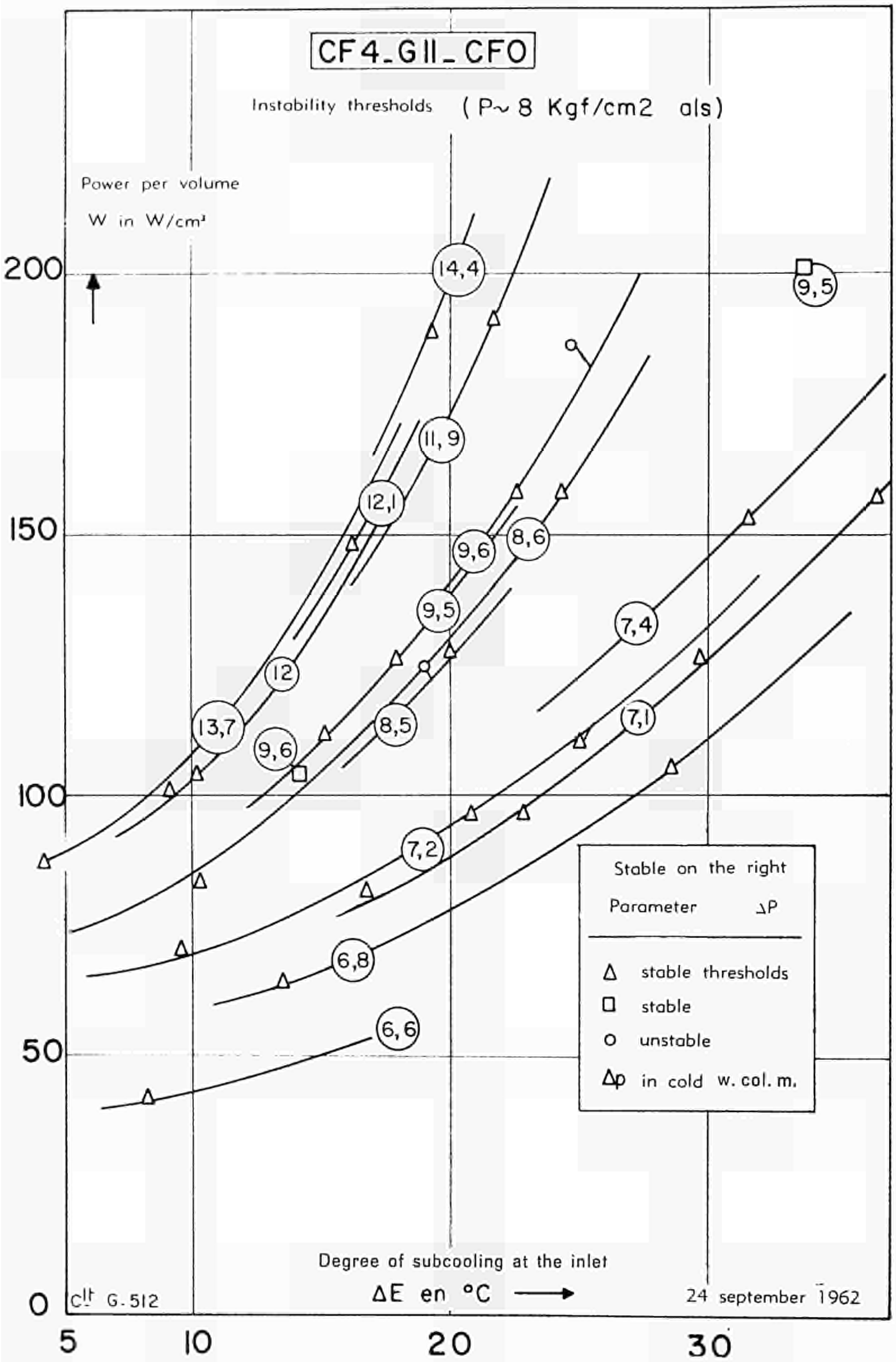
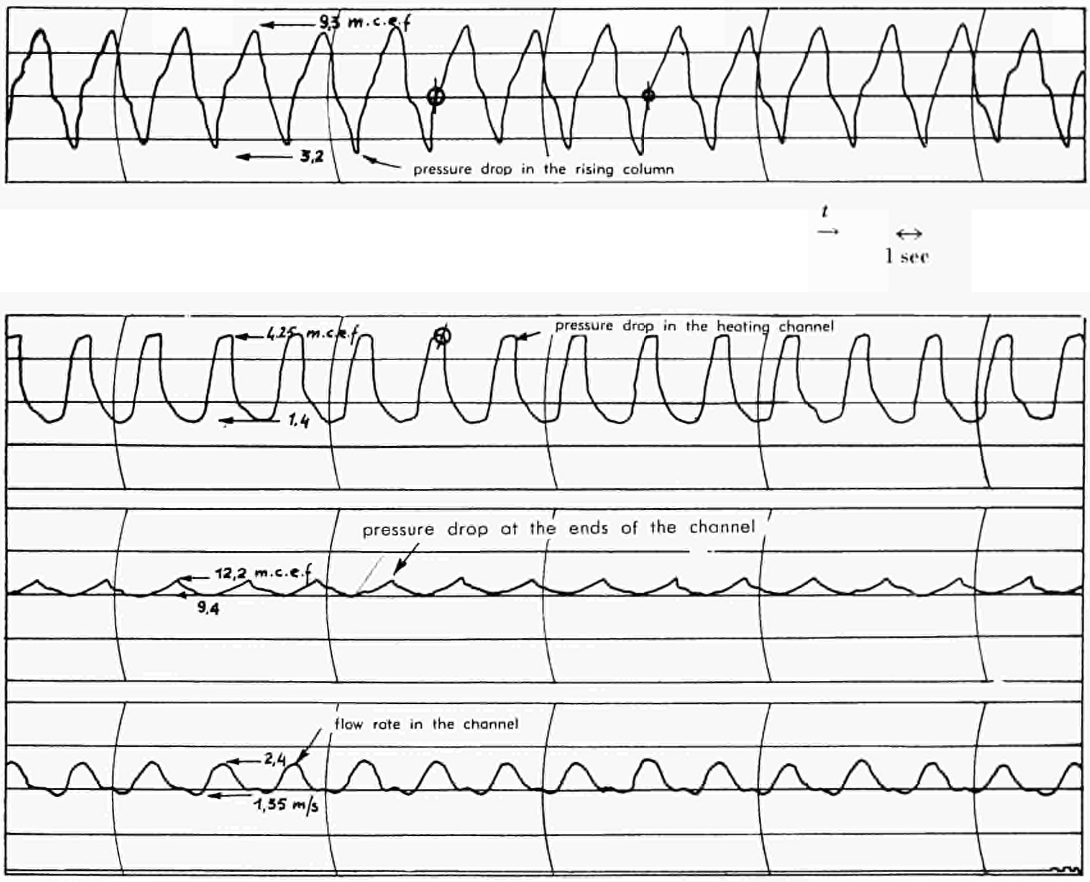

CF4_GII_CFO
Instability thresholds (P~ 8 Kgf/cm2 als)
Power per volume
W in W/cm³
200
150
100
50
0
14,4
11,9
12,1
9,5
9,6
8,6
9,5
12
13,7
9,6
8,5
7,4
7,1
7,2
6,8
6,6
Stable on the right
Parameter ΔP
stable thresholds
stable
unstable
Δp in cold w. col. m.
Degree of subcooling at the inlet
ΔE en °C
Clt G.512
24 september 1962
5
10
20
30

Figure 3



T: 1,7 sec
ΔP BP unstable
w ~ 190 w/cm³

CF 4 - G 11/FO
Experiment 15 ST Top 4
dated 13-7-61

Figure 4

e) EFFECT OF THE NATURE OF THE HEATING WALL

The previous experiments were carried out with an aluminium channel having a diameter of 6×8 mm.

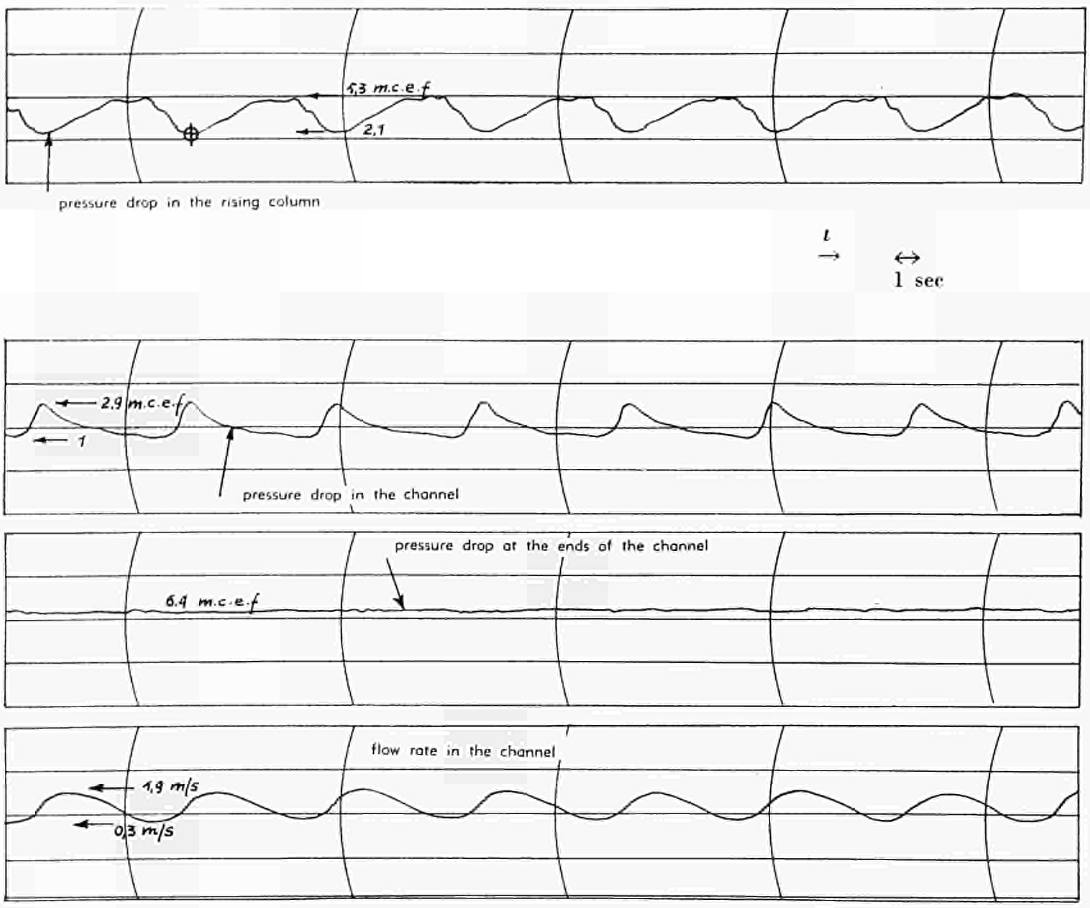
Experiments resumed with a stainless steel channel having a diameter of 6×7 mm revealed the following anomalies:

— disappearance of the oscillations at low steam quality and the occurrence, the parameters having approximately the same value, of slow burn-outs characterized by a low thermal flux (of the order of 10 W/cm²), a slow rise in the wall temperatures (of the order of 10° second), the rise levelling out at a limit value corresponding to a superheating of some hundreds of degrees C.

Moreover, the channel section superheated in this way may be relatively short and located any point between the ends (at least with low thermal fluxes).

— a slow burn-out phenomenon occurring almost immediately after certain oscillations at a high steam quality.

Several interpretations are possible; in particular we would mention the difference in longitudinal thermal conductivity between the two channels.



T: 3,4 sec
ΔP BP stable
w: 77 w/cm^3

CF 4 - G 11/FO
Experiment 12 ST Top 2
dated 23-6-61

Figure 5

3. A STUDY OF THE INSTABILITIES ON THE 1-ATMOSPHERE LOOP

The study was above all qualitative, the loop only being provided with fairly elementary measuring devices.

3.1. *With Natural Convection*

The main points noted were as follows:

a) under unstable conditions, slug flow with considerable coalescences in the riser;

b) recurrence of stability when the degree of under-saturation is reduced at the inlet or the power is increased;

c) the period seems to follow a linear decrease when the degree of subcooling decreases or when the power is increased;

d) if the length of the riser is decreased, at a given power:

— there is a reduction in the period in the region of the threshold,

— there is a reduction in the width of the instability zone,

— the low steam quality threshold occurs at a lower temperature;

e) Figure 6 shows the correlation existing between the flow rate and the volume of steam in the channel (measured from the separator level).

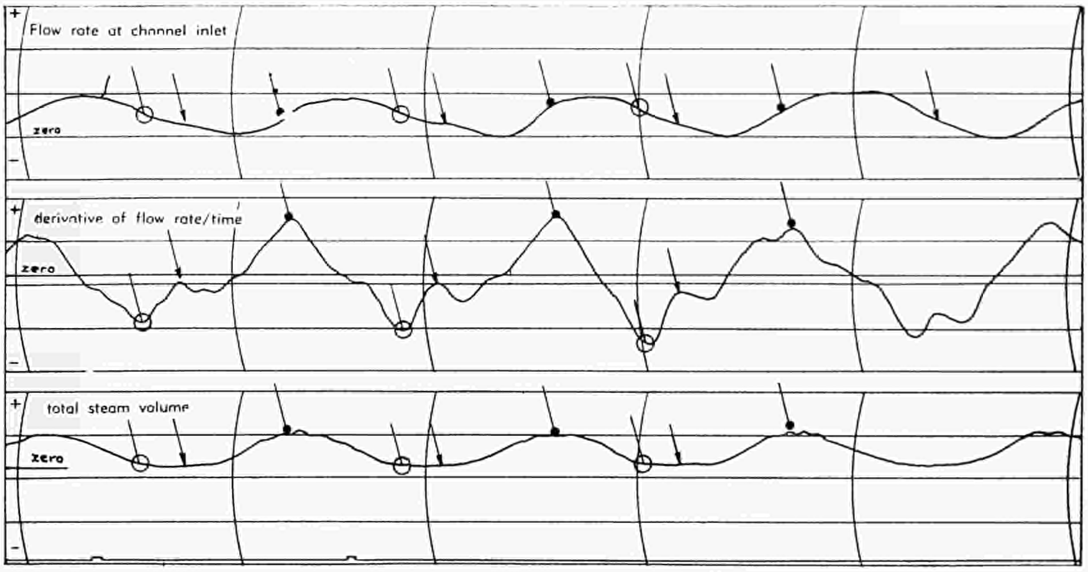


Figure 6
Recording on the glass loop. Top 1.

t 1 sec.

● beginning of vaporization: friction increases, average acceleration increases.

↘ end of vaporization: rapid pressure drop.

○ disappearance of steam: pressure remains constant

between ○ and ↘ : exponential decrease from M to SP; constant pressure.

between ↘ and ● : pressure rises rapidly.

between ● and ○ : pressure drops rapidly recondensation and evacuation into the separator.

It will be noted that with each oscillation there is a passage via a single-phase zone.

Moreover, the derivative of the flow rate versus time shows relative maxima and minima which are characteristic of the appearance and disappaearance of the steam.

3.2. *Forced Convection with By-pass*

The investigation was carried out with both a vertical and a horizontal channel. It was found:

a) that with a vertical channel oscillation is no longer possible when the riser is sufficiently short;

b) that oscillations are possible with a horizontal channel.

But they differ considerably from those obtained with a vertical channel:

— the periods appear to be lengthened,

— the flow is distinctly stratified,

— oscillation becomes very violent, with inversion of the flow rate at the channel inlet. The phenomenon seems to be related to chugging.

4. CONCLUSION

It can be seen that we are confronted with an extremely complex phenomenon.

Actually there seems to be a definite correlation between the instabilities and the burn-out, at least in certain cases. We would point also to the still somewhat mysterious part played by he wall and to the following two factors deserving of notice, viz.:

— the possibility of oscillation with a horizontal channel,

— the existence of two boundaries of a given geometry.

NOTE BY Mr. VILLENEUVE

A mathematical model for explaining the instabilities is now being studied at the laboratory.

In this model the instability would be due to a retarded effect of the flow rate on the pressure drop caused by elevation (weight of the emulsion column). The delay is connected with the transit time, in the riser, of a perturbation in the void fraction at the inlet.

In order to calculate this transit time use was made of the velocity of the continuity waves in the riser.

This velocity is close to the velocity of the liquid in the region of the pressure field investigated. The periods thus calculated are close to the experimental values.

With a horizontal channel a similar theory could be applied by postulating a retarded effect on the friction term.

Moreover, an interdependence was found between the continuity waves and the compressional waves which could cause blocking effects.

THE NUCLEAR VAPOTRON

Ph. DEMANGE, E. DOUGUET, J.D. LE FRANC
Compagnie Française THOMSON-HOUSTON, Nuclear Division, Bagneux, France

The results presented here were obtained within the framework of a study contract concluded between the European Atomic Energy Community (Euratom) and the three firms AEG, ALSTHOM and C.F.T.H.

We are concerned here with the adaptation of the Vapotron process to the field of boiling-water nuclear reactors.

The Vapotron process has been operated by the Compagnie Française Thomson-Houston for more than ten years. Discovered and developed by Mr. Beurtheret, it related essentially to electron power tubes and in particular to the cooling of their anodes by boiling water.

It is not within the scope of this review to give a detailed explanation of the operation and principle of this system. For this purpose it would be useful to consult the publications which have appeared on the subject (see literature references appended).

It can be said very briefly that a conventionally isothermal transfer surface (at least on average) is replaced by a surface provided with fins or corrugations on which a temperature gradient is set up (see figure 1).

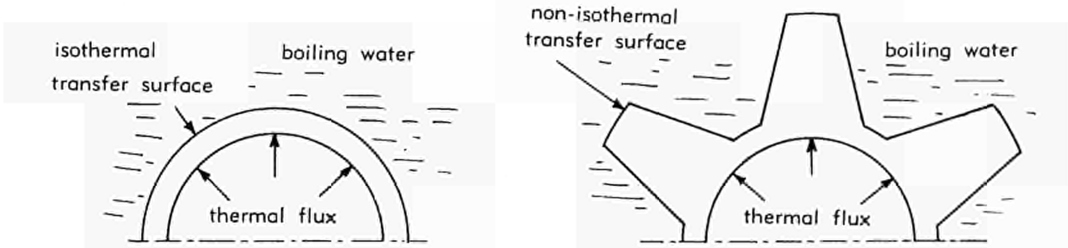


Figure 1

The hottest spots, which are those nearest to the heat source, can then attain elevated temperatures (often higher than the "critical" temperature) since they are stabilized by the cold spots located at the tips of these fins; the latter generally remain at a temperature which is scarcely higher than the saturation temperature of the water.

The effects produced by this gradient make it possible to obtain total thermal fluxes 2-4 times higher than those which could be extracted from isothermal surfaces under the same conditions. Furthermore—and this is a very important point—the operational reliability is greatly enhanced.

Figure 2 represents the thermal exchange law known as the Nuki-Yama curve. The heavy line shows the gradient which is set up on the transfer surface in the *Vapotron* phenomenon.

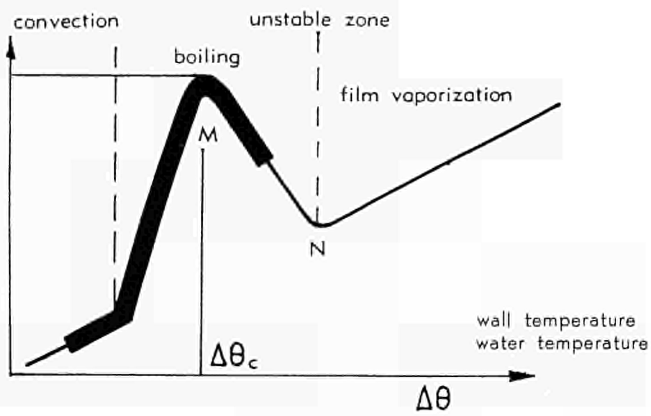


Figure 2

For transfer on an isothermal wall, a given wall temperature would be specified for a given flux. In the *Vapotron* case, on the other hand, at a given flux there is a continuous and stable temperature gradient along the surface of transfer.

It will be noted that the hottest spot can be situated either in the reputedly unstable zone or even in the film vaporization zone.

This effect is illustrated by the photographs in Figures 3 and 4, which show a sample consisting of a copper bar of rectangular cross-section. The upper and lower horizontal surfaces are thermally insulated. The thermal flux is injected horizontally into the sample and comes from the right-hand side in the photographs.

Figure 3 represents a fairly low flux. The three regions (unstable zone MN, turbulent and nucleate boiling zones, convection zone) are clearly visible.

Figure 4 represents a higher flux. The unstable zone has shifted to the left, the film vaporization zone has appeared, but the phenomenon has nevertheless been entirely stabilized by the sample's cold extremity, on which boiling occurs.

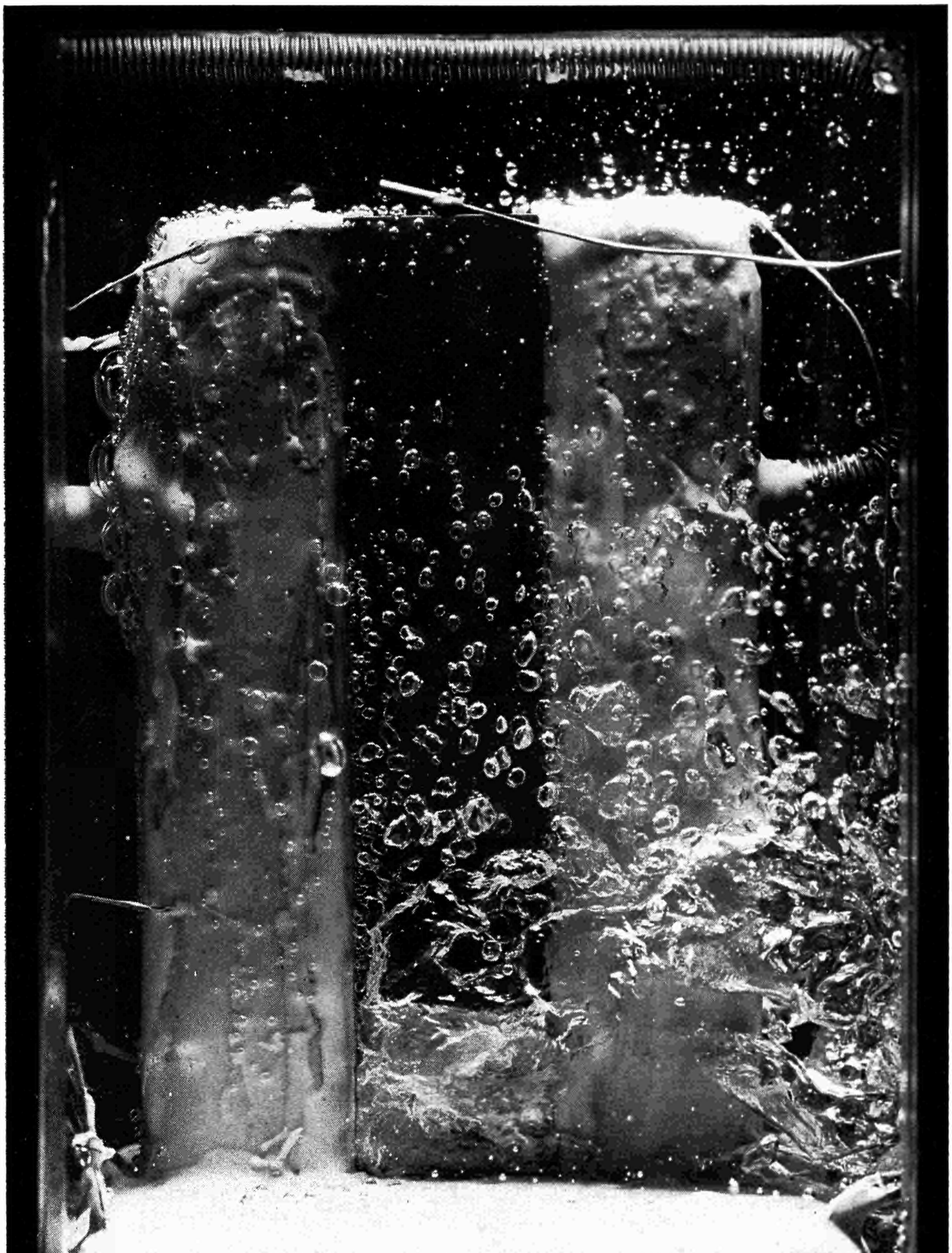


Figure 3

Figure 1

The problem we were required to solve under the terms of the study contract was to evaluate the conditions under which this process based on the use of copper at atmospheric pressure, could be adapted for use in the field of boiling-water nuclear reactors.

The first task was to replace the obviously "non-nuclear-material" copper by metals generally employed as fuel-cladding materials (Zircaloy or stainless steel), which unfortunately have a much lower thermal conductivity, and then to test these samples in boiling water at 70 atmospheres.

The general outline of our program is sketched out in the following paragraphs. This survey will at times appear to be cursory and of a somewhat "a priori" nature. For fuller information, reference can be made to the detailed progress reports which were compiled at various stages in the course of the study.

For the purpose of simplification, the tests related initially to elementary teeth operating under pool-boiling conditions, the horizontal surfaces of which were thermally insulated in such a way as to present a plane problem involving two dimensions only (see Figure 5).

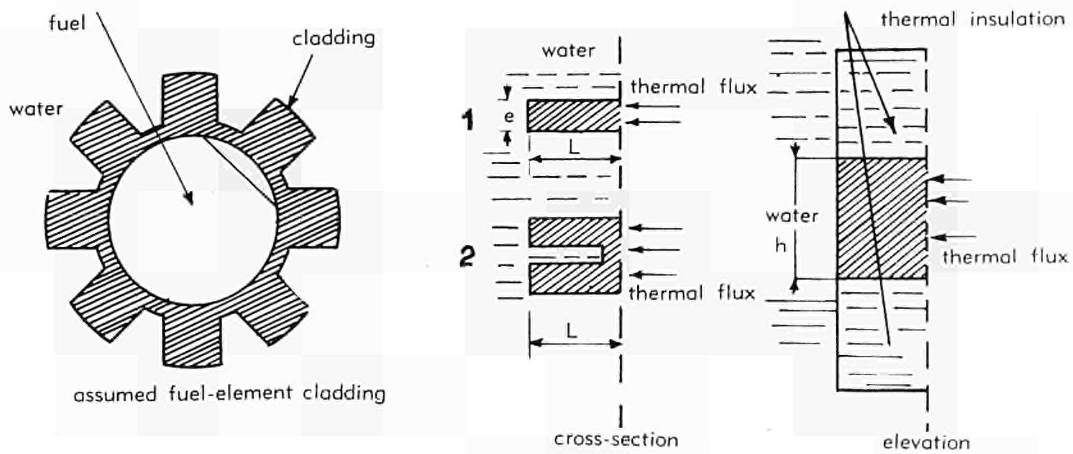


Figure 5

Test samples.

The samples were first tested without an intermediate channel (1) so as to eliminate any hydrodynamic effect, and then with a channel (2) in order to determine its effect.

It must be supposed that these insulated fins are subsequently applied around the entire circumference of a cladding, as illustrated in Figure 5.

After an initial series of experiments relating to various metals, we have now arrived at the following dimensions in the case of stainless steel:

Length	$L =$	3 mm
Thickness	$\delta =$	1.7 mm
Height	$H =$	20 mm

It will be seen that the parameters L and δ are geometrically compatible with a boiling-water reactor lattice.

These samples are fitted with thermocouples installed below the surface, or further inside the metal, by means of which it is possible to trace the temperature gradient along the axis for a thermal flux applied at the base.

Using several test rigs, we were able to work at various pressures (from 1 to 70 atm.). It should be borne in mind that all the tests mentioned were carried out without forced water circulation, that is to say either with natural circulation, at low speed, or most frequently with pool boiling.

Figure 6 shows the overall flux dissipated by a single stainless-steel sample (*referred to the base surface*) as a function of the temperature difference between the base and the water for various pressures.

It will be observed that at about 70 atm it is possible to dissipate 800 W/cm^2 and more without any cessation of boiling at the extremity of the sample.

We can define the Vapotron effect in this manner: when the cold extremity of the sample reaches the temperature of point M (see Figure 2), boiling ceases to occur on the surface, which is surrounded by a blanket of steam and which is operating wholly within the film vaporization zone. This is the total burn-out phenomenon. It should be noted that this state can be stable and may not entail the destruction of the sample despite the considerable temperature rise involved.

It was interesting to attempt to define the sectors of the plans ϕ/A, $\Delta\Theta$ where the Vapotron effect is maintained and those where total burn-out occurs.

Figure 7 gives the temperature difference measured at the tip of the sample as a function of the pressure for various thermal fluxes. The same graph also shows a curve produced by Cichelli and Bonilla which expresses $\Delta\Theta_C$ as a function of the pressure; this agrees well with our experimental results in the low-pressure zone. The region situated inside this curve therefore corresponds to the existence of the Vapotron effect, since boiling is still taking place at the cold extremity.

Figure 8, which was deduced from Figure 7, gives the thermal flux corresponding to the transition from Vapotron effect to total burn-out as a function of the pressure. As we did not carry out any experiments at pressures exceeding 70 atm, those portions of the curve extending beyond that figure represent extrapolations.

We can thus ascertain the maximum flux likely to be obtained with a simple rectangular sample. It will be observed that this maximum is about 1 kW/cm^2 at 70 atm. Since the transfer surface/base surface ratio is about 4.5, this corresponds to an *average* thermal flux of approximately 220 W/cm^2 on the transfer surface.

In order to bring in the study of hydrodynamic parameters, we subsequently conducted experiments involving a dual sample (type 2, Figure 5) with a channel width of 1 mm.

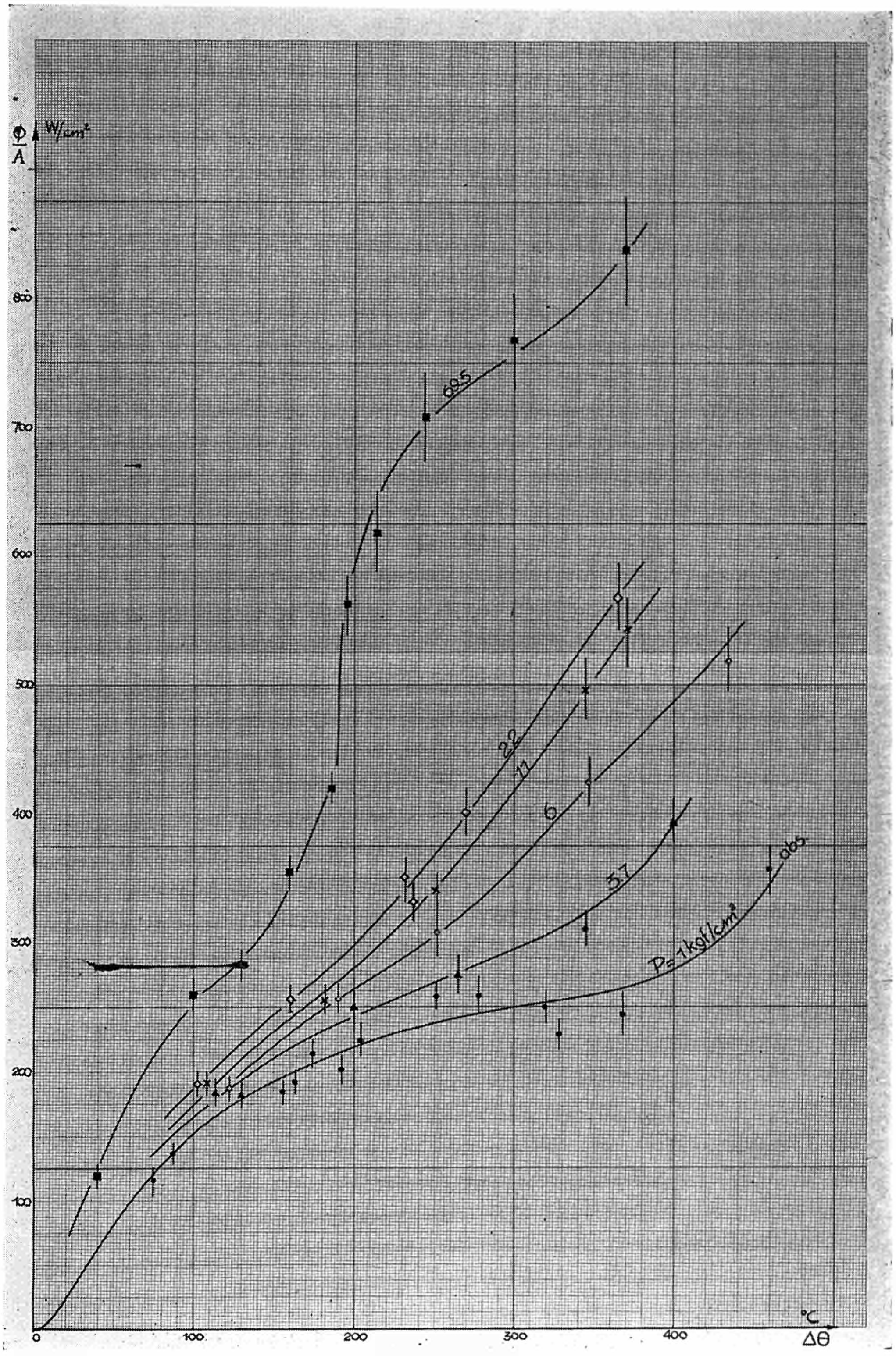


Figure 6

Thermal flux in the base as a function of the base/saturation temperature difference.

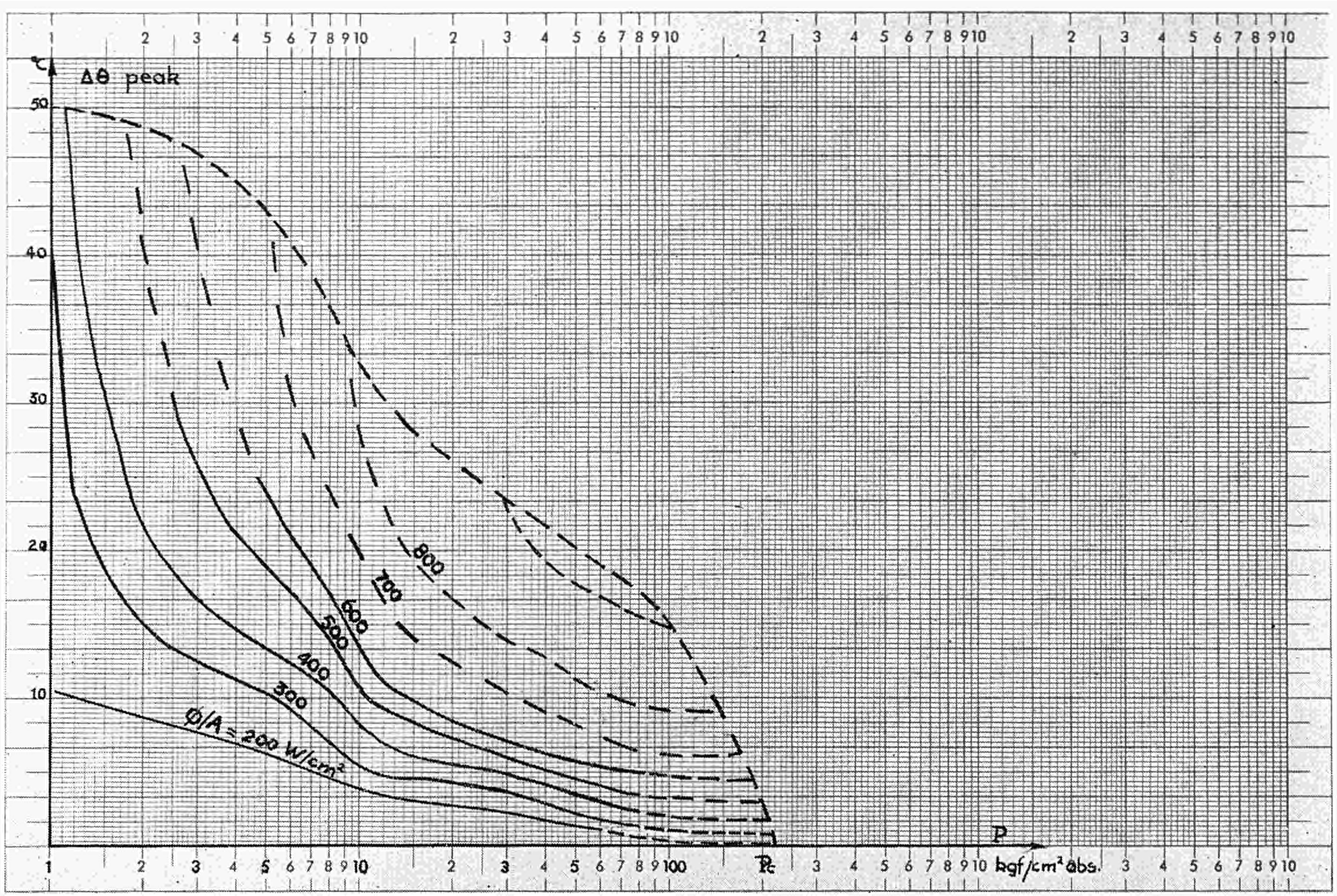


Figure 7

Stainless steel sample 3 × 1.7 mm.

Top/saturation temperature difference as a function of pressure (parameter: flux in the base).

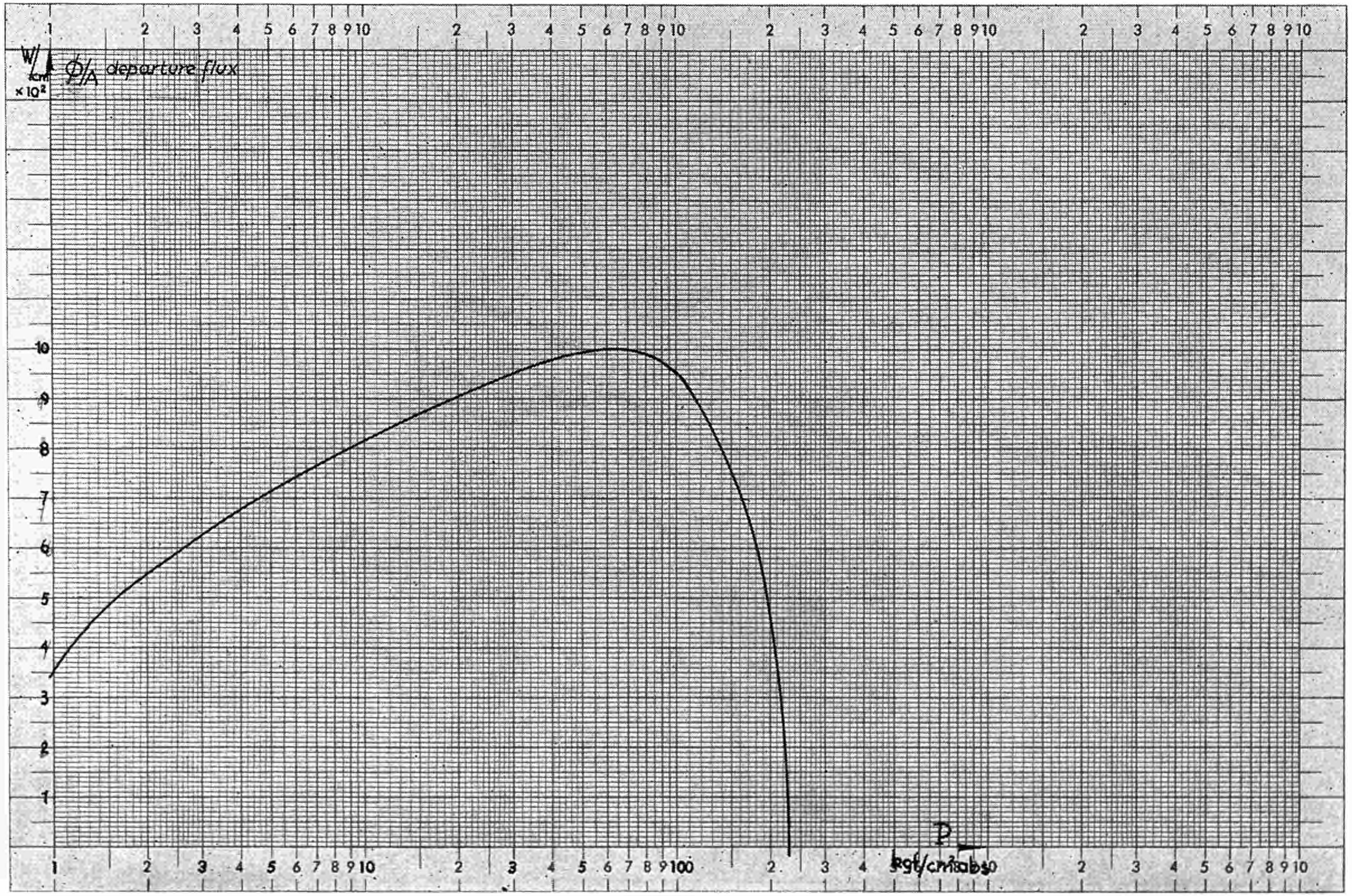


Figure 8
Departure from vapotron effect heat flux as a function of pressure.
Stainless-steel sample.

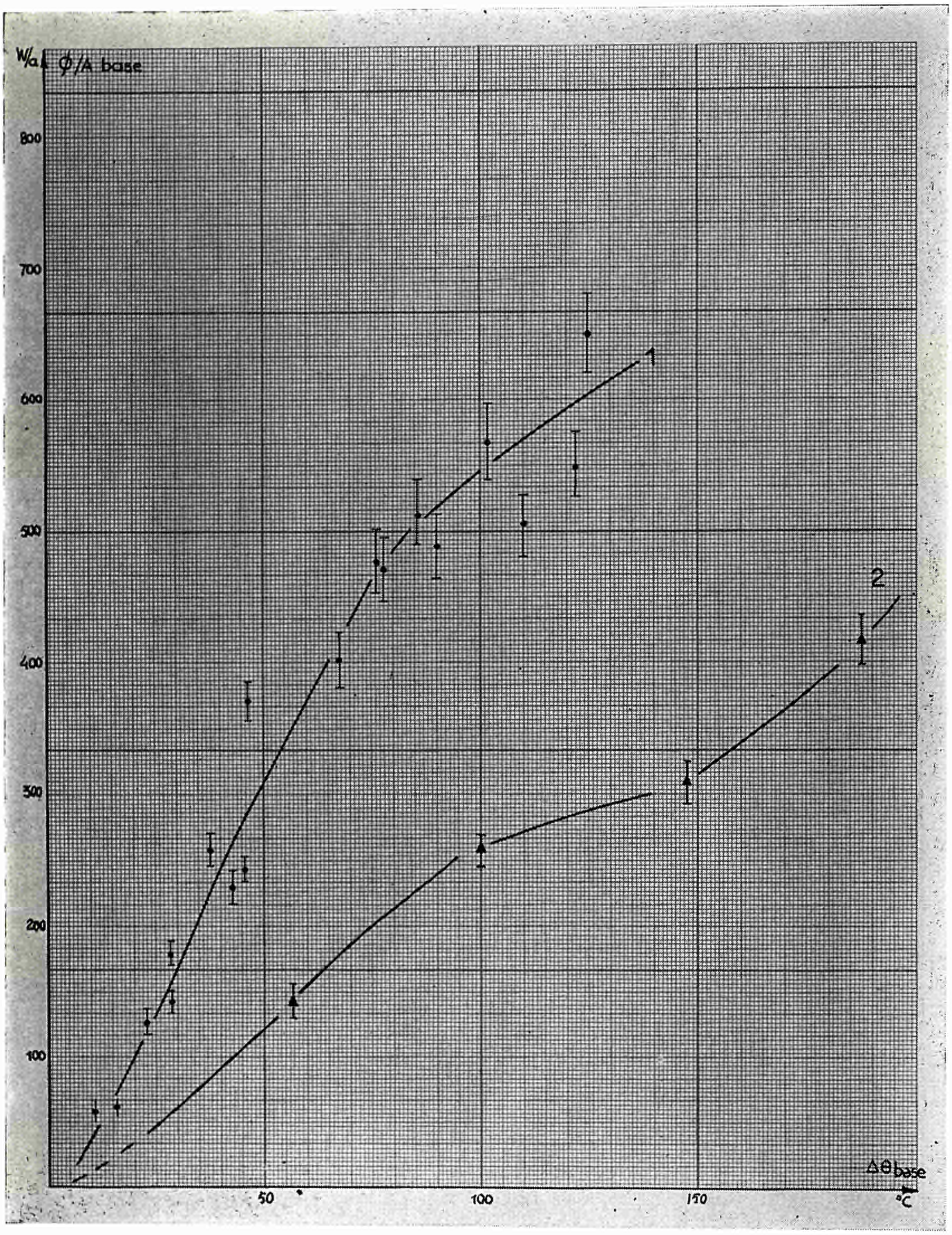


Figure 9

Comparison between:

— *a sample consisting of two 3 × 1.7 mm stainless-steel fins* (1) *and*
— *a sample consisting of a single 3 × 1.7 mm stainless-steel fin* (2).

The results recorded are shown in Figure 9. The favourable influence of the channel is immediately apparent. Moreover, this entirely corroborates the experience gained elsewhere in the field of electron tubes.

At equal flux the base temperature—referred to the water temperature—is approximately half, the hydrodynamic parameters giving the advantage of better heat-transfer conditions.

Tests are currently in hand with a view to establishing the optimum values for the three dimensions (length and thickness of the fin and width of the channel).

Now that this first series of experiments with shapes which were purposely kept simple have proved the validity of the system under pressure and with metals of low conductivity, a second series is being conducted with more realistic structures simulating a nuclear fuel element: a vertically finned cylindrical cladding is heated at its centre by a graphite rod. Figure 10 illustrates one of these samples.

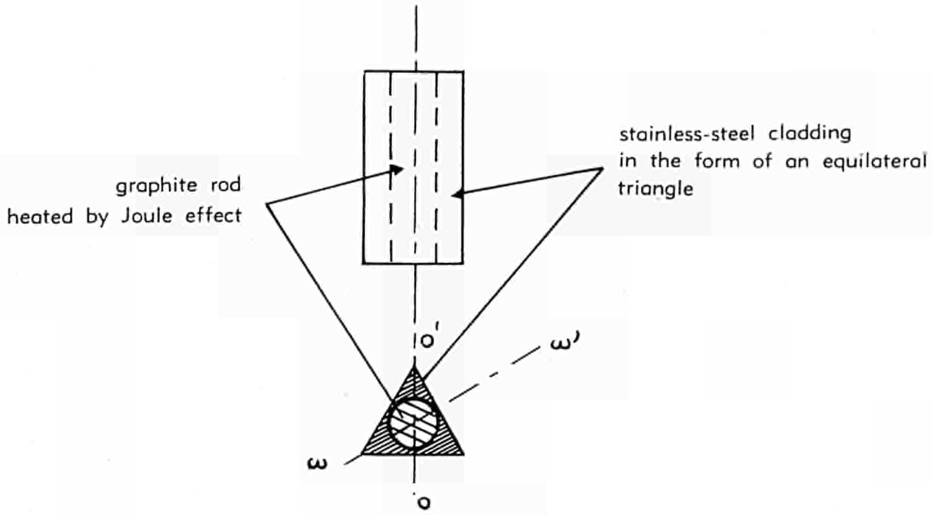


Figure 10

Figure 11 shows several photographs of this sample taken according to the thermal flux dissipated in the graphite.

Photographs A, B and C represent increasing fluxes which lead to a marked Vapotron effect; they were taken along the 0,0' axis.

Photographs D, E and F, which were taken along the ω, ω' axis, represent higher fluxes; it can be seen from photograph E that only a very small zone, situated at the extremity of the ridges, is not included in the boiling.

At a flux of 200 W/cm^2 it was possible to allow such a cladding to reach red or even yellow heat at its centre, and to remain in that state for several minutes, without causing destruction of the sample. On the other hand, a short cylindrical tube heated in the same way was destroyed very rapidly at a flux of 90 W/cm^2.

In case it is believed that this triangular type of cladding constitutes a very poor approximation, from both the thermal and the neutronic standpoint, a comparison of the above two figures will suffice to show the interest of the Vapotron system.

The continuation of our program provides for "realistic" experiments in a test loop. This loop, which is at present under construction at the heat-transfer laboratory of Société *Alsthom* de St. Ouen (Seine), should be in operation by about mid-1963.

It is planned to test rods equipped with Vapotron claddings of various shapes. These rods, which are about 50 cm long, will be tested under forced circulation. In the initial phase, involving a single rod, the total electric power may reach 150 kW.

I should like, at this juncture, to point out that another important part of our program aims at attempting to explain this thermal efficiency, which may seem surprisingly high.

First of all we worked on the elaboration of mathematical models which could subsequently be handled on digital computers. These models can be constructed either on the basis of our experimental results or of analogical studies conducted, for example, with the rheographic tank.

It quickly became apparent, however, that the necessary means—which also happens to be the ultimate objective of the study—is to obtain a better understanding of the so-called NukiYama curve. The results that can be found in the literature on the subject are very widely scattered, even in the case of experiments with isothermal surfaces. Our impression is that the heat-transfer law can "a fortiori" be substantially modified when the transfer takes place on a surface which causes a steep thermal gradient.

Tests relating on the one hand to the study of local temperature variations (either immediately below the metallic surface or in the boundary layer) and on the other hand to the analysis of fast motion films, seem to us to constitute a possible approach to the problem.

We believe that in our case the peak of the heat-transfer curve, which therefore corresponds to the maximum efficiency, is much flatter than in the case of isothermal transfer. In particular, during the analysis of films shot with an ultra-high speed camera it was observed that, instead of escaping vertically, the steam produced in the film boiling zone moves laterally across the metallic surface as though under the influence of suction, before escaping into the zone of normal boiling. As a result the surface shows a relatively wide zone which proves to be the site of extreme turbulence—undoubtedly very favourable to the heat transfer—and which corresponds to the maximum efficiency.

In conclusion, let us say that the primary object of our investigations was to demonstrate the efficiency of the Vapotron process in the operation of boiling-water reactors. From another point of view, however, it is necessary to emphasize the possible value of a heat transfer study on zones normally held to be unstable or dangerous, which are stabilized as a result of the Vapotron effect and consequently lend themselves to a systematic investigation.

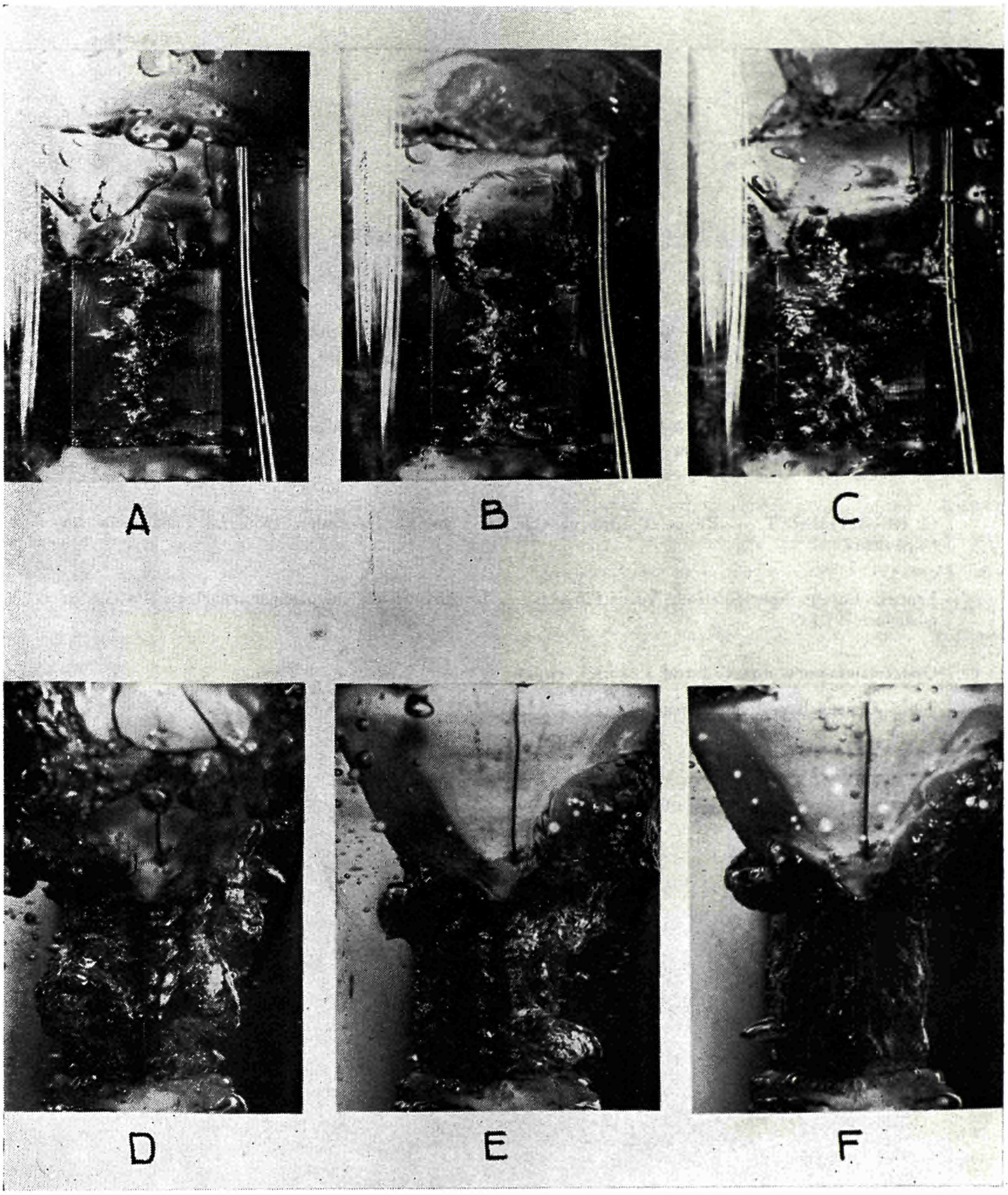


Figure 11

LITERATURE REFERENCES

C. BEURTHERET. — *Les processus de vaporisation et le Vapotron* (lecture given during the heat-transfer conference organized by the Institut Français des Combustibles et de l'Energie) June 1961.

C. BEURTHERET. — *L'ébullition à flux imposé sur paroi non isotherme* (lecture given during the VIIth Hydraulics Conference organized by the Société Hydrotechnique de France, 1962).

C. BEURTHERET. — Revue Technique C.F.T.H., No. 24, December 1956. *La Technique des Vapotrons.*

French Patent Specification No. 1 126 414. — Improvements in nuclear reactors. Patent of addition 799.139.

Quarterly reports nos. 1, 2 and 3 - Final report

Research contract Euratom/AEG - ALSTHOM - CFTH - No. 001-61-6 A R F.

ANALYSIS OF VARIOUS BURNOUT DATA (LITERATURE RESEARCH) AND COMPARISON OF AVAILABLE DATA

H.H. AGENA

Maschinenfabrik Augsburg-Nürnberg, AG, M.A.N., Nürnberg, Germany

The critical analysis of burnout research results is part of our research contract. The first difficulty in such an analysis is the definition of the burnout point. Every laboratory has its own definition which depends more or less on the measuring technique applied. A second difficulty is that the test conditions are not given exactly enough. As a third difficulty it should be mentioned that many of the data published are incomplete. In view of this it is very often impossible to calculate the hydro-dynamic and thermodynamic properties and states exactly.

It therefore seems to be difficult to decide on the feasibility of one or the other burnout correlation.

At present one can divide the different types of general burnout correlations into three groups:

1. Correlations established on a purely empirical basis.

2. Correlations based on criteria of the stability of the two phase separation layer in the unstable and stable film-boiling ranges and dimensional analysis of these criteria.

3. Correlations on the basis of dimensional analysis of the conditions of nucleate boiling.

The first group contains the well-known JENS-LOTTES [10] and BETTIS [1] correlations as well as the CISE [11] correlations and others. Generally, these correlations are valid only for a small range of hydrodyanimc and thermodynamic conditions. The correlations found are reliable from an engineer's point of view but it is dangerous to extrapolate from these correlations for other conditions than those on which they are based.

The second group of burnout correlations concerns those formed on the basis of instabilities of the two phase separation layer in the unstable and stable film boiling ranges. The main work was done by Zuber, Tribus and Westwater [2, 5] as well as Kutateladse [3]. Though this is not necessarily valid for forced convection on vertical heated surfaces Iwaschkewitsch [4] uses it for correlating data under these conditions.

Referring to Russian test results he gets a deviation of $\pm$ 25 to 50%.

The third group of burnout correlations is based on dimensional analysis of nucleate boiling phenomena. Papers in this connection are published by Griffith [6, 7], and

Rohsenow and Griffith [8]. The correlation which Griffith [6] proposes is as complicated as that of Iwaschkewitsch. Griffith reaches a deviation of ± 33% for a similar range of variables as Iwaschkewitsch [4].

Some time ago we started a rough comparison of test results which seemed to be taken under similar conditions. At present we are preparing a programme on a digital computer to compare test results on different general bases.

Comparing the results of Aladyev [9] and co-workers with the Jens-Lottes [10] correlation for sub-cooled liquids one can see that the Jens-Lottes correlation is conservative. There is a small difference of only 5 to 10% between Jens-Lottes correlation and Aladyev data at 180 atm.abs. and water velocities of 3 to 5.5 m./sec. For higher pressure Aladyev's data are smaller than values calculated from the Jens-Lottes correlation. For 110 atm.abs. the Aladyev data are 17 to 36% higher than the data calculated from the Jens-Lottes correlation

The curves obtained by plotting the critical heat flux against absolute pressure according to the Jens-Lottes equation, the Bettis equation, and Aladyev's data respectively show entirely different trends. Comparison of Aladyev's data with Bettis's correlation give relatively small differences only at pressures near 140 atm.abs. between 2-7 m./sec. which seem to be of the order of measuring errors. With velocities under 2 and over 7 m./sec. the differences are large.

It is difficult to compare data in the quality region because of hydrodynamic instabilities. As shown in the papers of Aladyev [9], Iwaschkewitsch [4] and Kutateladse [3] there are considerable influences of pulsations on burnout heat flux in the low quality region between 0 and 50% quality.

If there is a stable non-pulsating flow one gets a continuous decreasing heat flux when exit quality increases. With pulsations in the test channel Aladyev [9] and Silvestri [11] found very small burnout heat fluxes in the low quality region.

LITERATURE

[1] R.A. DEBETOLI, S.J. GREEN, B.W. TOURNEAU, M. TROY, A. WEISS. — USAEC Report WAPD-188, Oct. 1958, Pittsburgh, Penna.

[2] N. ZUBER, M. TRIBUS, J.W. WESTWATER. — Int. Develop. in Heat Transfer, Aug. 61, p. 230.

[3] S.L. KUTATELADSE. — Reprint of a paper presented at the conference on heat and mass transfers in Minsk, USSR, June 5 - 10, 1962, (will be published shortly in Journal of Heat and Mass Transfer).

[4] A.A. IWASCHKEWITSCH. — Teploenergetika, 1961, Vl. 8, No. 10.

[5] ZUBER. — USAEC-report AECU-4439, June 1959.

[6] P. GRIFFITH. — ASME Paper No. 57-HT-21, 1957.

[7] P. GRIFFITH. — USAEC-report: WAPD-TM-210, December 1959.

[8] M.M. ROHSENOW, P. GRIFFITH. — AIChE-Meeting, Louisville, Kentucky 1955.

[9] I.T. ALADYEV, Z.I. MIROPOLSKY, V.E. DOROSCHUK, M.A. STYRICOVTCH. — Int. Developments in Heat Transfer/Int. H.T. - Conf. Boulder Colorado, USA, Aug. 61.

[10] W.H. JENS, P.A. LOTTES. — USAEC-Report ANL 4627, 1951.

[11] M. SILVESTRI. — Int. Developments in Heat Transfer/Int. H.T. - Conf. Boulder Colorado USA, Aug. 61.

PREPARATION OF AND PRELIMINARY TESTS FOR BURNOUT MEASUREMENTS

F. MAYINGER
Maschinenfabrik Augsburg-Nürnberg, AG, M.A.N., Nürnberg, Germany

The work carried out under the contract concluded between US-Euratom and M.A.N. is concerned with measurements of the critical heat flux (burnout) in boiling water.

A major portion of the preliminary work went into the design and construction of the test loop which is rated for pressures up to 200 atm. and temperatures up to 360 deg.C. at a maximum volumetric flow of 15 cu.m./hr. The total installed electric capacity of the heating arrangements is 550 kW.

The water and steam temperatures in the circuit will be measured by means of steel-jacketed thermocouples providing a pressure-tight connection to the tanks and the piping. The surface temperature of the test channels will be determined by a method suggested by Buchberg [1] with three thermocouple wires—two of nickel and, between these, one of nickel chromium—welded direct on the uncooled surface of the heated channel with a distance of 0.5 to 1 mm. between the individual wires and connected by a Wheatstone bridge. The temperature of the cooled surface will be computed from the readings of these "three-legged" thermocouples by numerically solving the differential equation for the temperature distribution in steady-state temperature fields with heat sources, taking into account the temperature dependent coefficients of thermal conductivity and electric conductance. As these thermocouples can only cover a temperature field spotwise the change in electric resistivity of the channel wall will be used as an additional parameter in determining the impermissible temperature rise preceding burnout. The absolute pressure will be measured by means of piston-type pressure gauges developed by Siemens. For measuring minor pressure differentials under steady-state conditions we are at present developing a differential pressure manometer which at absolute pressures of more than 200 atm. will enable us to measure pressure differentials with an accuracy of $\pm$ 0.1 mm. The steam quality of the mixture at the inlet of the test channel will be computed from the heat balance of the preheater upstream of the test channel. The steam quality at the oulet will be determined by measuring the separated water and steam flows downstream of a steam separator. We are also working on a measuring device which will enable us to determine the density of the steam/water mixture at the bur-out point from the change in impulse rate of a gamma-ray source.

The test program proposed by us and laid down in our contract with Euratom covers measurements of the critical heat flux under conditions of boiling in individual channels of circular, annular and rectangular cross-section and hydraulic diameters of

5 to 15 mm. The ratio of heated length to diameter will be varied between 50 and 100. The length of the unheated inlet section is to be varied from 0 tot 700 mm. The thermodynamic parameters extend from 20 to 200 atm. and from 60 deg.C. subcooling to 10% steam at the inlet. The cold water velocities range from 0.5 to 5 m. per sec.

The object of our investigations is to contribute to a clarification of the influence which the flow regime has on the burnout phenomenon. According to the available literature the influences of hydraulic diameter and heated channel length are expected to be small. Tests made by Aladyev [2] and Berkowitz [3] showed, however, that the burnout heat flux is influenced by the inlet conditions. This influence shows itself in instabilities of the absolute pressure and the mass flow.

As the tests carried out by Berkowitz et al. at CISE were mostly made with high steam qualities we asked ourselves to what extent this influence of the inlet conditions still exists at low steam quality down to sub-cooled boiling. For this purpose we propose to vary the inlet conditions without, however, producing a swirl.

In view of the progress made to date it would appear that there are justified hopes of commissioning the test loop in February 1963.

LITERATURE

[1] BUCHBERG. — *Studies in Boiling Heat Transfer*, COO-24 March 1951.

[2] L.T. ALADYEV et al. — *International Developments in Heat Transfer*, Boulder, Colorado USA, Part II, p. 237.

[3] L. BERKOWITZ et al. — Report R-27, Cise October 1960.

ADAPTATION OF VORTEX FLOW TO A BIPHASE LIQUID GAS MIXTURE

Cl. MOUSSEZ
S.N.E.C.M.A., Nuclear Section, Suresnes, France

1. INTRODUCTION

1.1. The studies carried out by S.N.E.C.M.A. under Euratom-USA contracts since 1960 have been directed at the improvement of heat transfer in boiling water reactors by means of the vortex flow principle.

The thermal flux in these reactors is restricted by heat conduction inside the fuel element and the maximum temperature in the centre of the fissile material, on the one hand, and by the critical heat flux to the wall on the other.

This latter value is the limit for which the temperature difference between the wall and the liquid increases up to a point where the heating element is destroyed by melting.

This phenomenon is bound up with the conditions governing the flow of the biphase mixture, but any action which tends to carry the steam away from and the liquid towards the wall may serve to retard its occurrence.

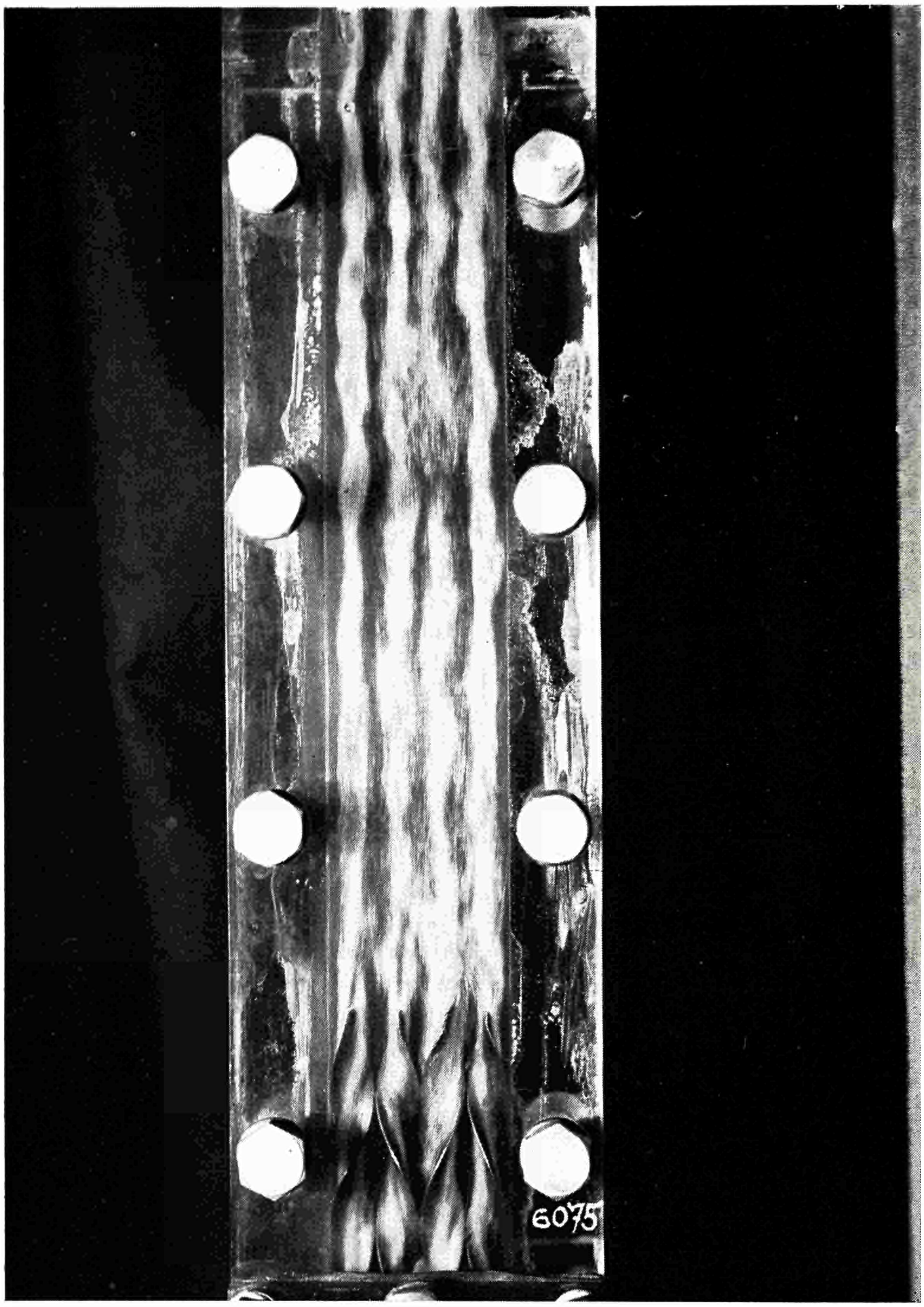
A number of research projects have demonstrated that this action could be carried out by rotating the liquid, but the test conditions did not correspond with those actually obtained in a boiling water reactor.

The tests carried out by S.N.E.C.M.A. are aimed at demonstrating the improvements brought about by the application of the vortex flow principle in flow and geometry conditions identical with those prevailing in reactors.

It was of particular importance to show the compatibility of an assembly of associated unit vortex flows induced by twisted tapes.

Figures 1 and 2 show, in preliminary tests, where the rectangular or annular test section is traversed by a water-air mixture, the path followed by and the distribution of the biphase mixture as well as the hydrodynamic compatibility of the unit vortices.

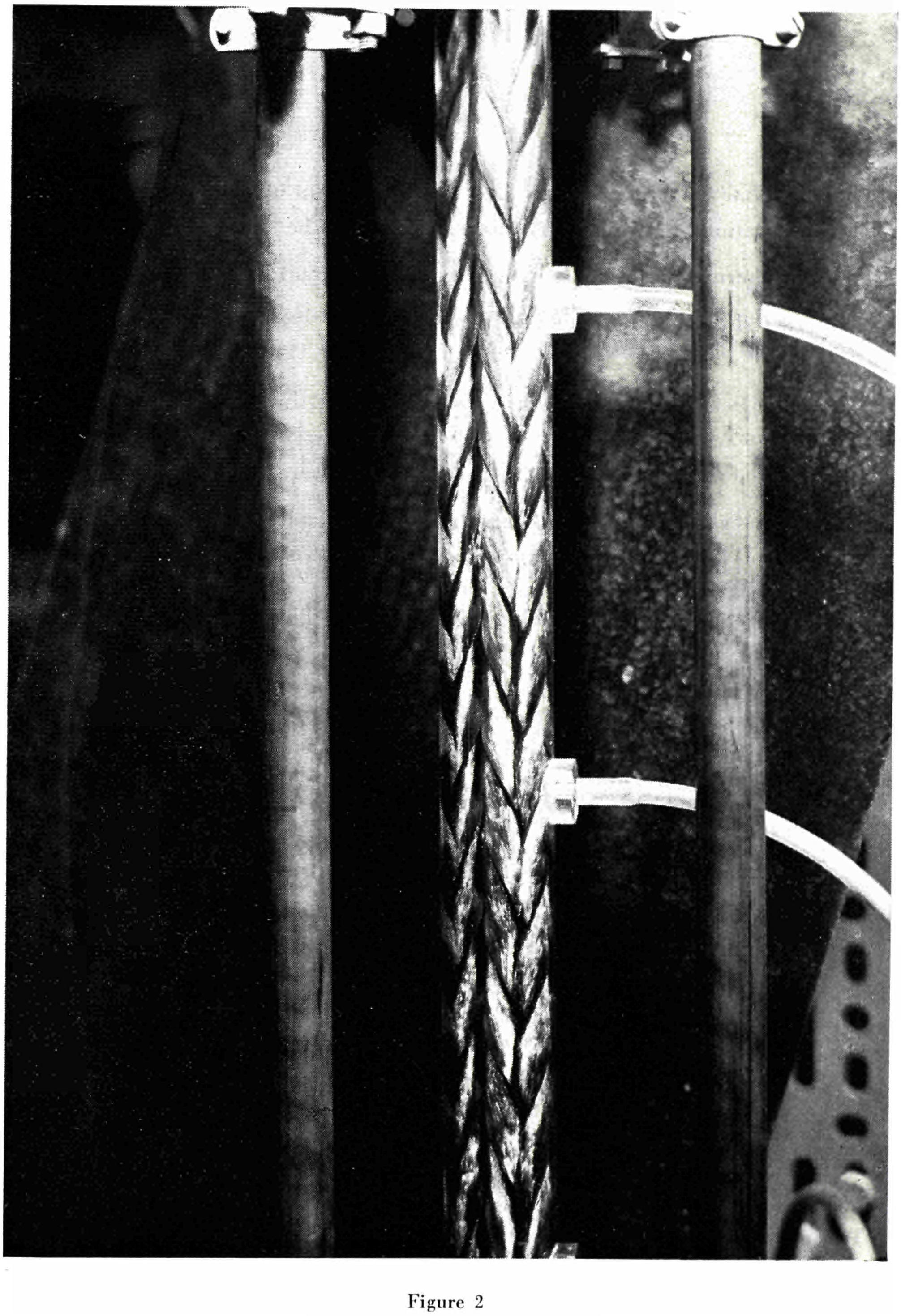
In both cases we can see that the light phase has collected in the axis of the spirals and on Figure 1 that the vortex flows are maintained after the device used for imparting a rotary motion to the liquid has been cut out.



↑
FLOW

Figure 1

Vortex flow in a rectangular channel.



↑
FLOW

Figure 2

Vortex flow in an annular channel.

1.2. *Types of Test*

The tests were concerned with a comparison of straight and spiral biphase flow, the three parameters used to arrive at the correlations being:

— the burnout flux

— void fraction

— pressure drop.

These parameters are necessary for the drawing up of a BWR draft design.

The installations in which such correlations may be obtained are of two types:

— a non-thermal installation with a semi-open circuit;

— thermal installations with closed-circuit loops.

1.3. *Test Sections*

1.3.1. The following geometrical configurations were tested:

— circular cylindrical canals - length 800 mm, diameters 10, 20, 30 mm;

— annular channel - length 800 mm, diameters 11/31 mm;

— "realistic" channel, simulating on scale 1 a rod-type square-pitch fuel element section: rod dia 14 mm

lattice pitch 19 mm

In this case, the interior of the test section is made up of 4 active rods (heated or unheated depending on the type of test), the outside boundary being:

— either rod shape (representation of an inside zone of a real fuel element);

— or with plane faces (representation of an outside zone, peripheral rods and walls of the box of a real fuel element).

In the thermal tests the electric power is supplied by Joule effect:

— to the tube constituting the circular channel (internal flow);

— to the inside tube of the annular channel (flow outside the heating tubes);

— to the 4 inside tubes of the "realistic" section (flow outside the heating tubes).

1.3.2. The twisted tapes are defined by their reduced semi-pitch y, the length necessary for a rotation of 180°, expressed as a multiple of the width of the band.

The circular test sections are fitted with an internal spiral of constant width, the annular test section with six constant-width spirals and the four-rod section with nine spirals which can either be at a tangent to the rods (constant width) or take up the space between the rods (evolutive width).

Inspected from the end, the constant-width spirals are said to be circular, while the evolutive spirals are defined as cruciform.

2. THERMODYNAMIC TESTS

The study of the water-air mixture in the neighbourhood of atmospheric pressure (1-5 bars) makes it possible to obtain the void fraction correlations and the biphase pressure loss factor.

2.1. *Test Installation*

The loop has a semi-open circuit:

— water-air supply - mixer - test section - separator - water recuperation into an upper tank - water returned into mixer via a feed pump.

2.2. *Measurements*

2.2.1. Measurement of flow rate of the two fluids and of pressure drop along the test section (pressure points on the rods and on the adjacent part of the channel).

2.2.2. The average void fraction is determined over the test section by measuring the average speed of the liquid phase by briefly injecting an appropriate tracer. If the outside of the test section is transparent, there are two possibilities for detecting the transit of the injected material by means of the two measuring points, one being located at the inlet and the other at the outlet of the test section.

The tracer is, in this case, conductive and coloured (salt + nigrosine).

The first detection system is optical (light beam passing through the transparent channel—detection by photoelectric cells) while the second method is electrical (electrodes under voltage flush with the channel walls and giving a signal when the electrolytic tracer passes through. If it is not possible to have a transparent wall (4-rod test section) the electric method alone is used.

Figures 3 and 4 illustrate the signals obtained without spirals with and without air for a circular transparent test section and a four-rod test section (Figure 4 shows the pattern followed by the signals as a function of the steam quality and in particular the appearance, in the case of high qualities, of almost pure signals having a form similar to those observed for water flow alone. This makes it possible to determine whether the conditions corresponding to the liquid film have been established.)

In a transparent mock-up, comparison between the results of the two detection methods and, generally speaking, at zero quality, comparison between the water flow rate measurement by the tracer method and the venturi method show a fluctuation of less than 5%.

It should be pointed out that experiments are now in progress in which the void volume is obtained by gamma measurement (source used Cs 137 - activity 1 curie), and comparison with the tracer method indicates that it will be possible to keep the fluctuations within the 5% margin.

2.3. *Results*

Variation in the parameters:

flow rate	0.2 to 4 m/s
mass quality	0.2 to 25%

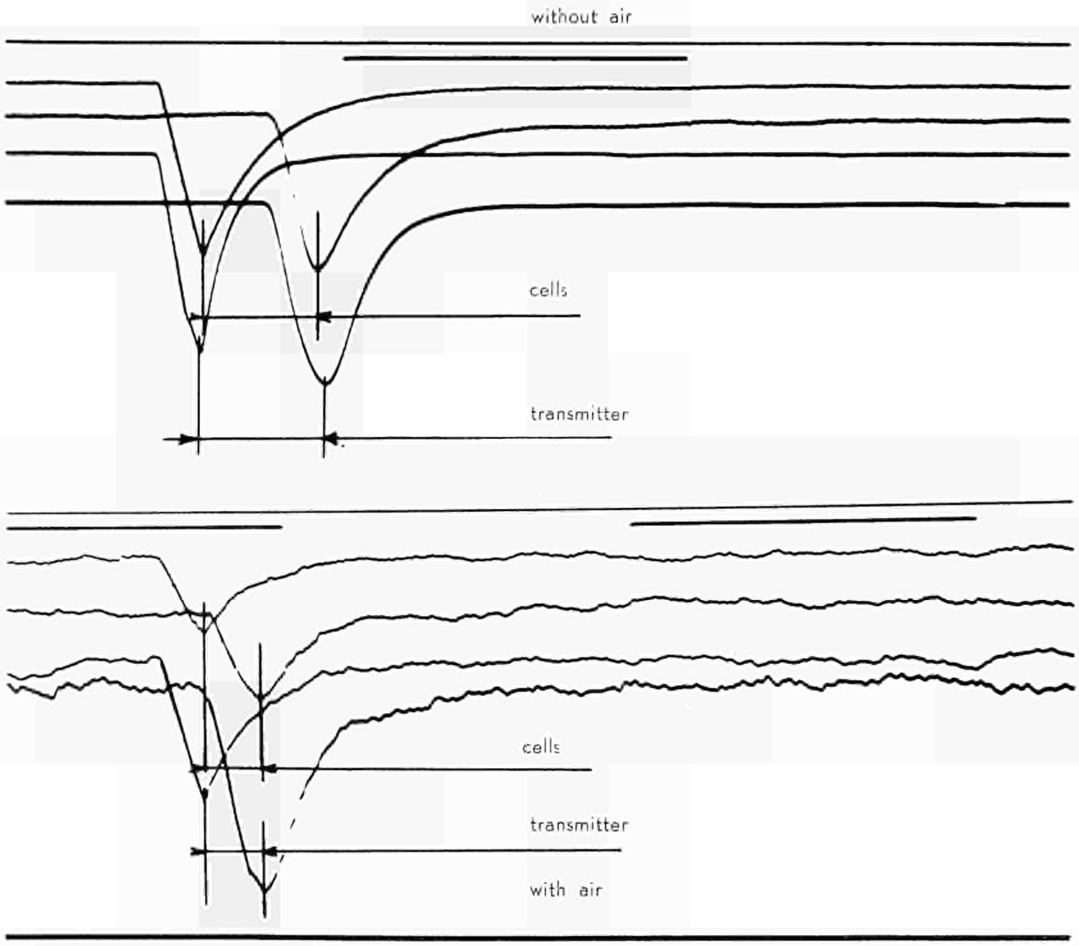


Figure 3

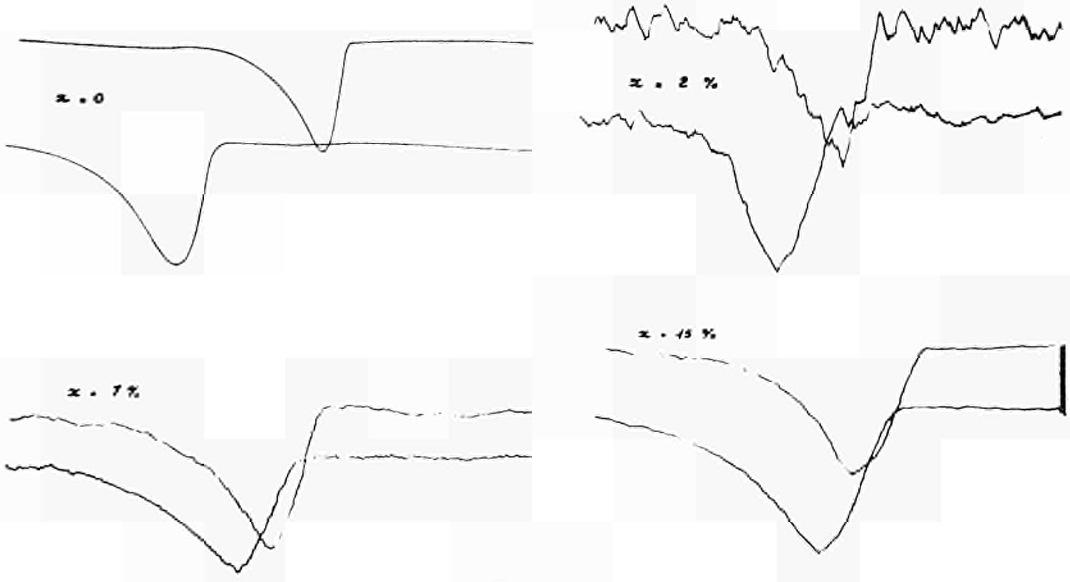


Figure 4

4 rods test section — $V_{l_0} = 0{,}5\ m/s$.

Generally speaking, the results continue to coincide fairly closely with the Martinelli-Nelson curves corresponding to atmospheric pressure. This applies to all configurations, even to those involving twisted tapes.

Furthermore, the degree of coincidence appears to be substantially better when we examine the spiral flow patterns induced by the bands.

The correlations using the friction coefficients both with and without spiral bands and the comparisons with the Martinelli-Nelson curves for the results obtained in the circular and annular test sections (effect of hydraulic diameter, flow rate, reduced half-pitch of the spirals) are set out in EURAEC reports Nos. 54 and 144.

Figures 5, 6, 7 and 8 illustrate, for certain experiments carried out on the four-rod test section, the void fraction and the friction pressure drop coefficient, with and without spiral bands, as a function of the Martinelli-Nelson parameter. The full curve represents, on each figure, the results obtained by these authors.

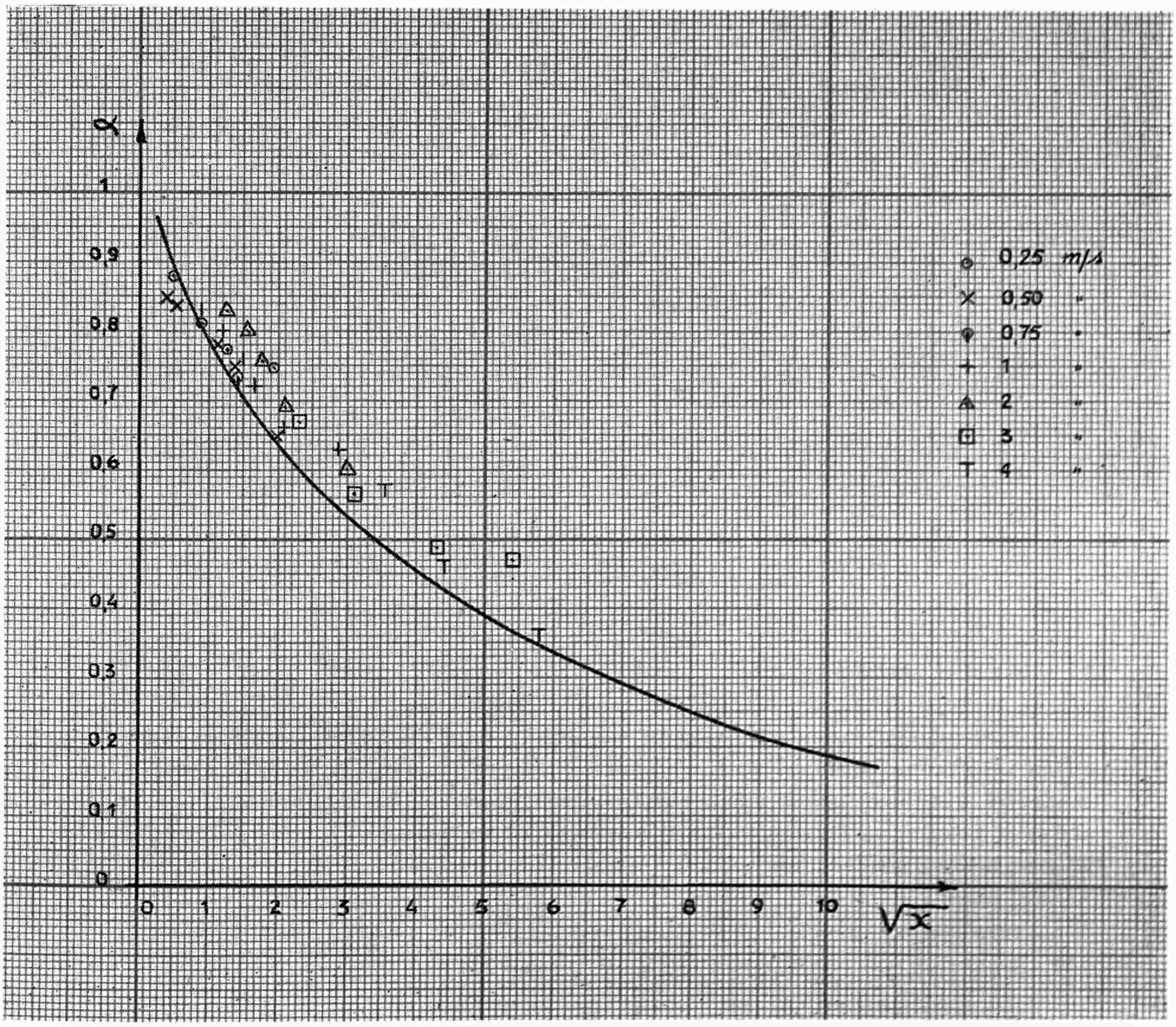


Figure 5

4 rods test section - Straight flow - Void fraction.

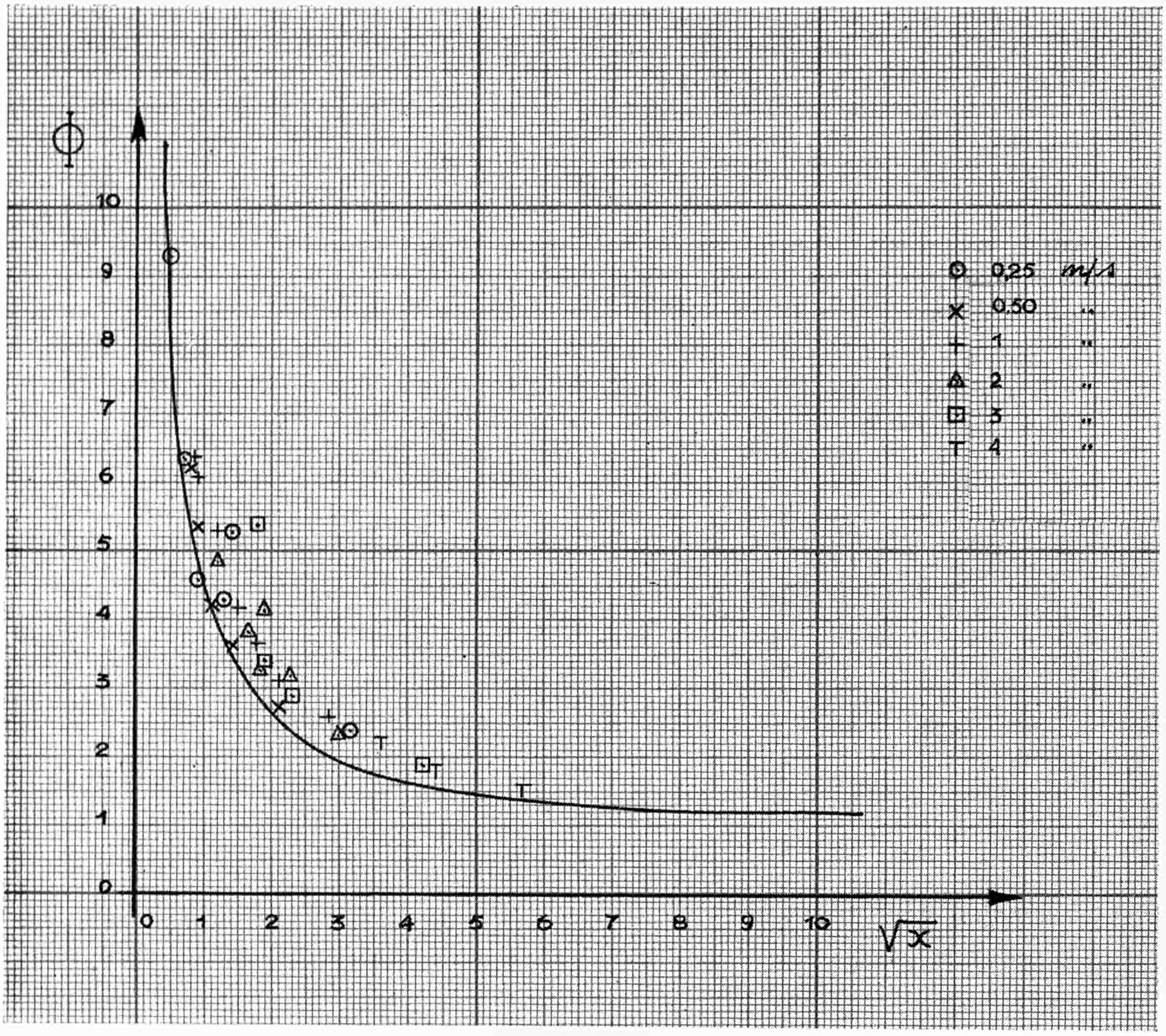


Figure 6

4 rods test section - Straight flow - Pressure drop factor.

3. THERMAL TESTS

Particular attention will be paid here to the results obtained from tests conducted on an installation equipped with a closed circuit or loop for which the pressure at the condenser is in the neighbourhood of atmospheric pressure. The high-pressure loop, which makes it possible to reproduce the thermodynamic conditions of a boiling water reactor, is now being installed. Its general features will be discussed below.

3.1. *Installations*

3.1.1. *Low pressure loop* (Figures 9 and 10)

Pump (maximum flow rate 7 kg/s, pressure head 15 bar)

Flow rate measurement section: venturis and turbines

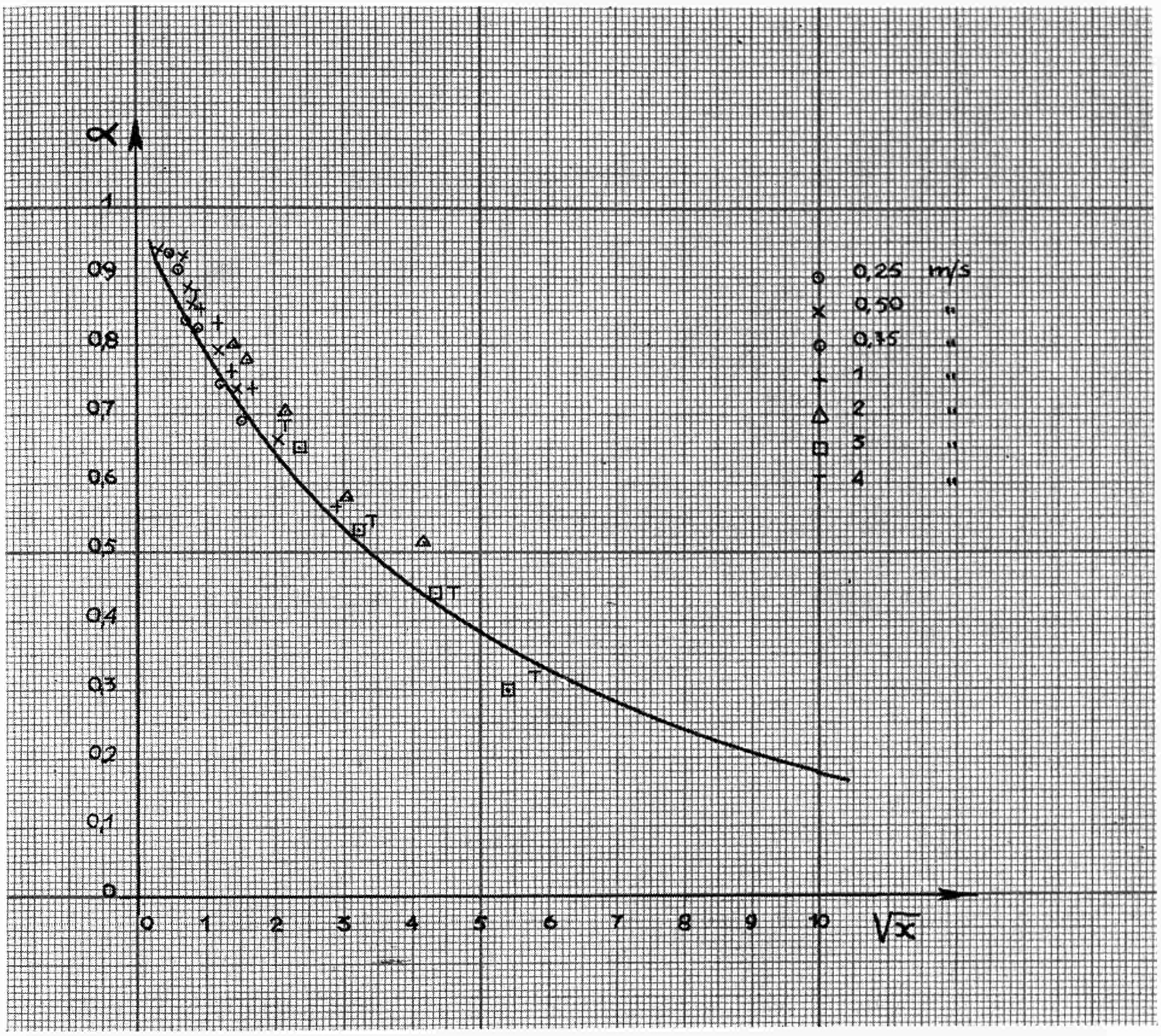


Figure 7
4 rods test section - Circular twisted tapes y = 3 - Void fraction.

Heater 25 kW

Cooler 25 kW

Test section 300 kW

Separator

Water condensers (atmospheric pressure, 100°C)

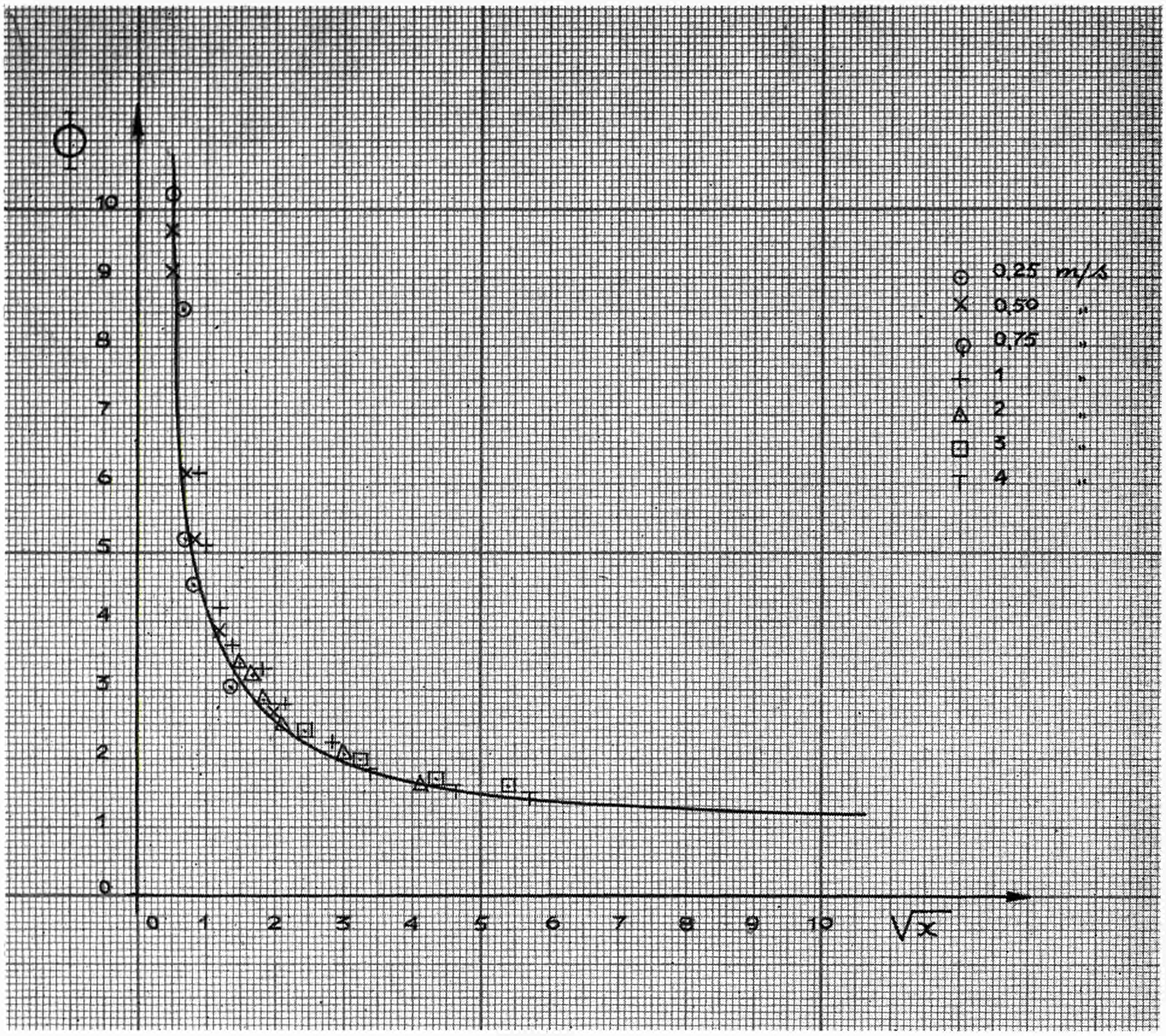


Figure 8
4 rods test section - Circular twisted tapes y=3 - Pressure drop factor.

3.1.2. *High pressure loop* (Figure 11)

Pump (maximum flow rate 11 kg/s, pressure head 20 bar)

Flow rate measurement section: turbines

Heater 200 kW

Test section 800 kW

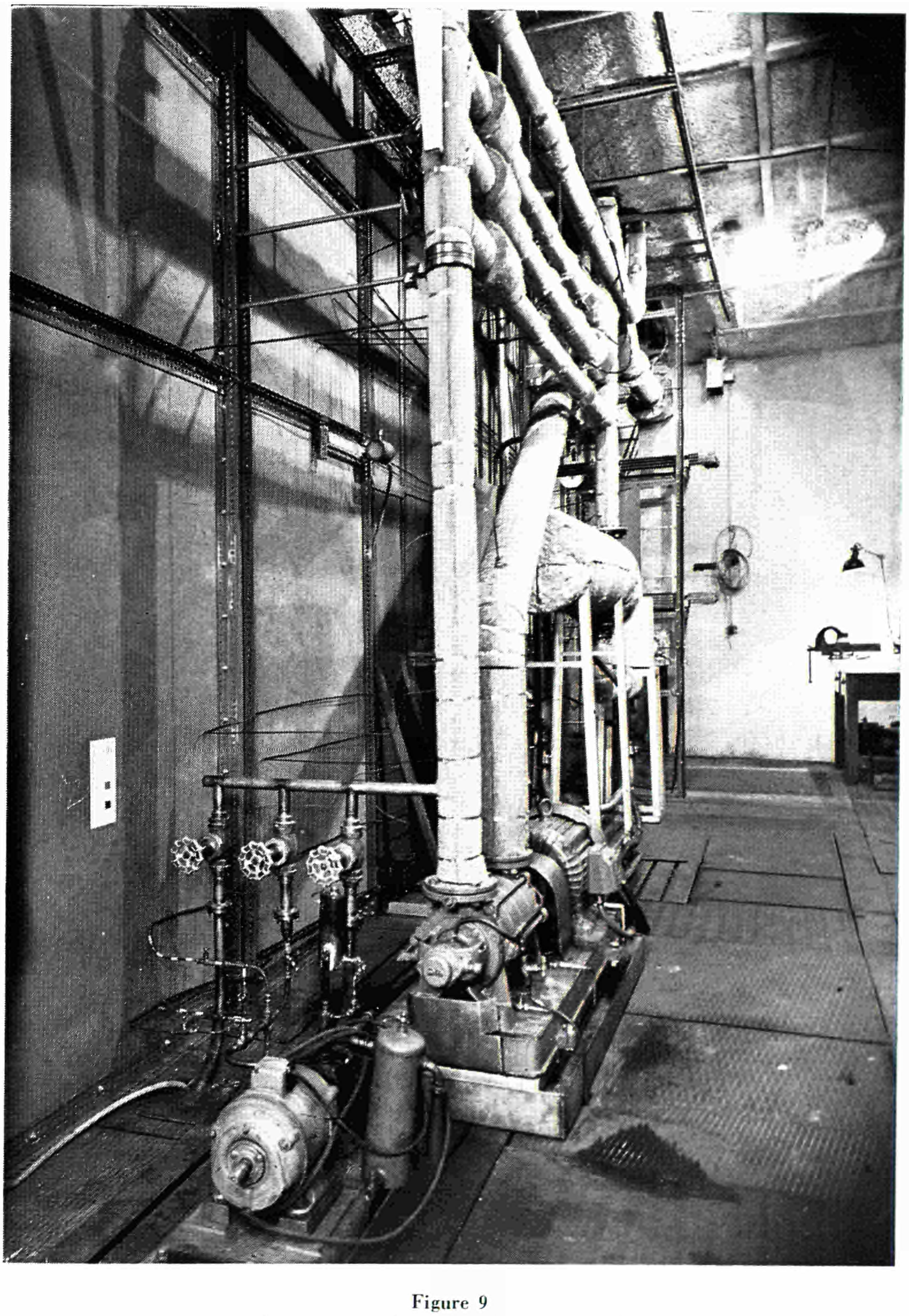


Figure 9
Low pressure loop view from pump side.

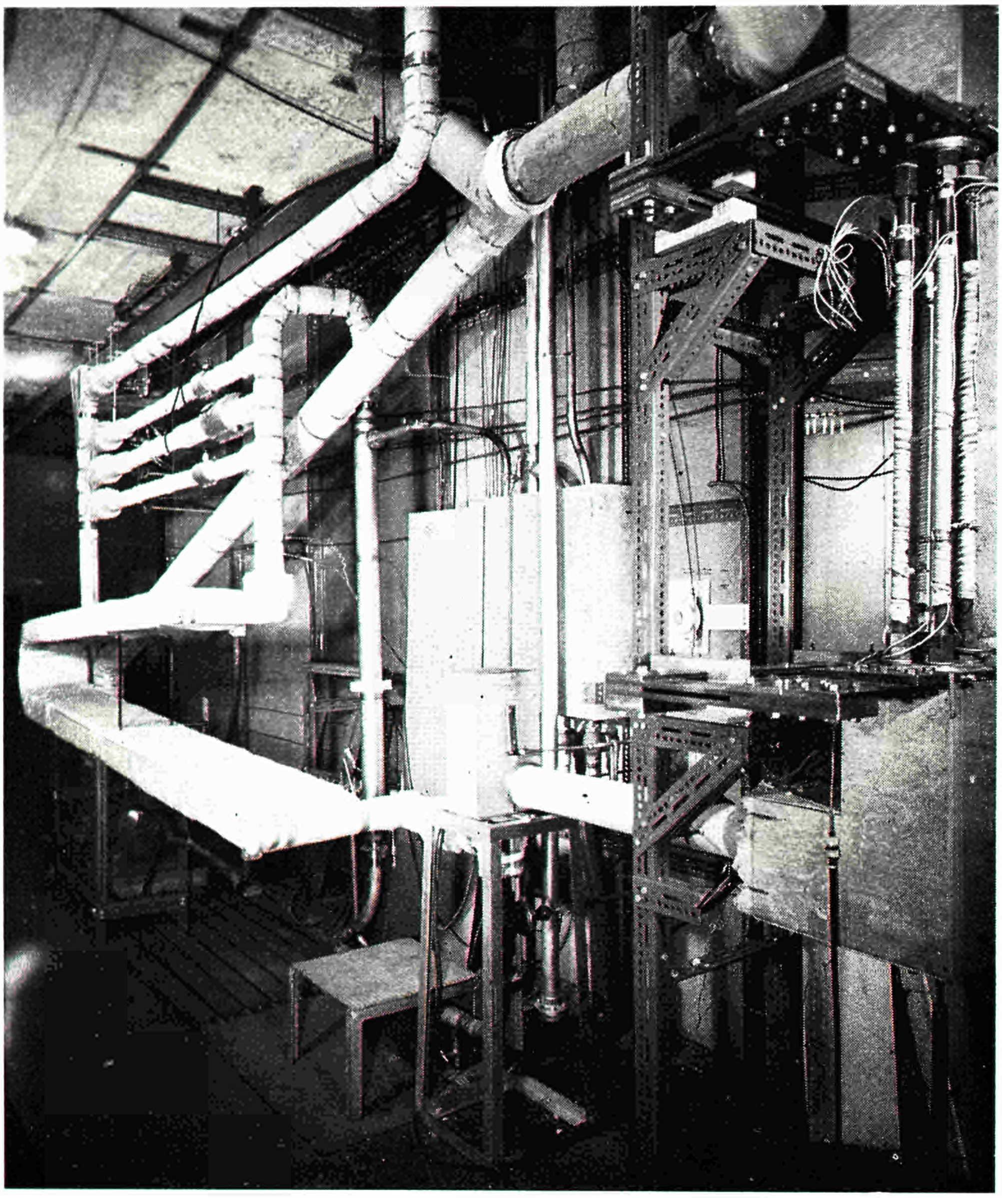


Figure 10
Low pressure loop view from test section side.

Separator

Aerocondenser set at 70 bar (285°C)

Cooler 300 kW

3.2. *Measurements*

3.2.1. Flow rates (pressurized lines of loops).

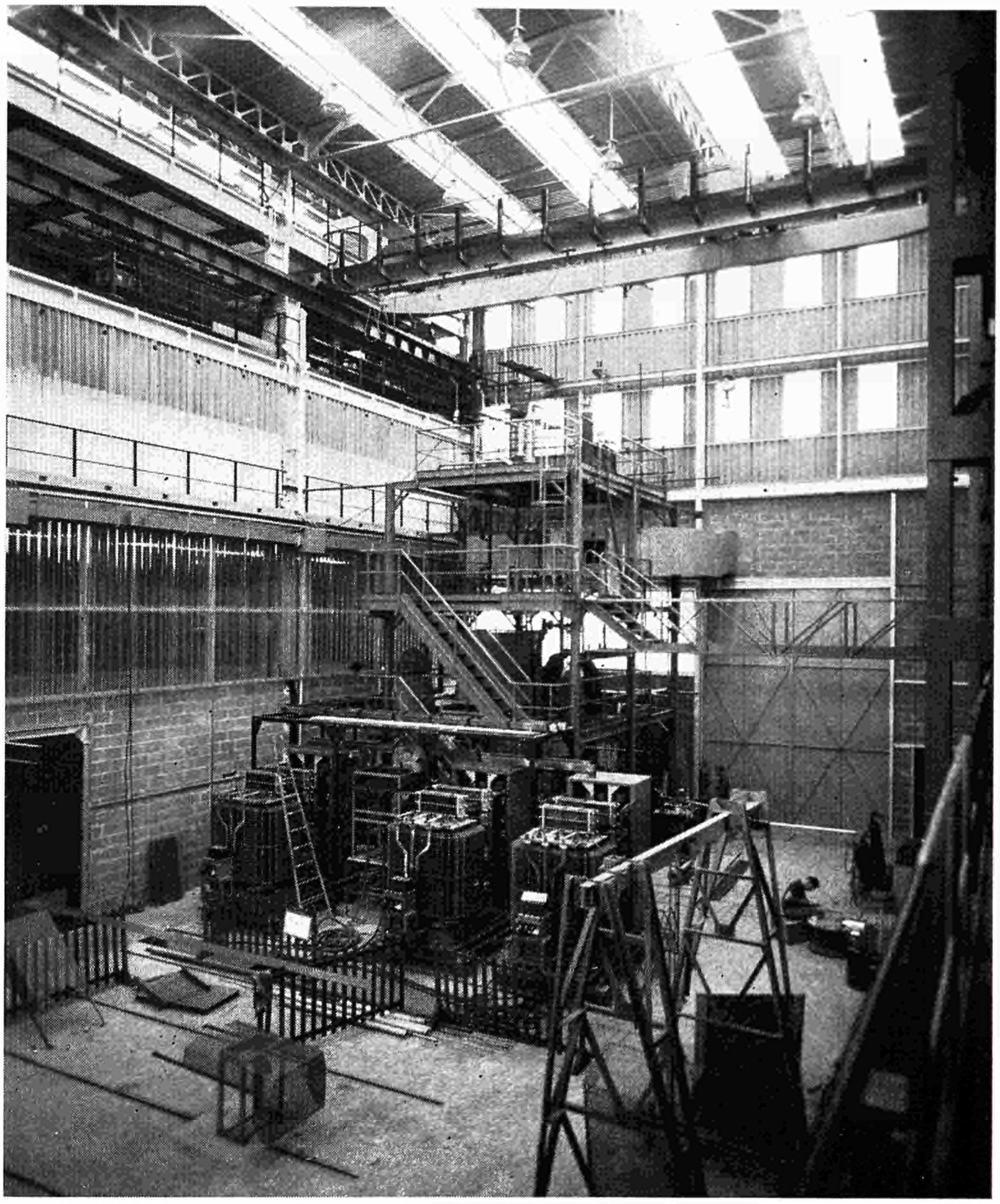


Figure 11

High pressure loop - Ground level: pumps, flowmeters, power supply.
1st floor: Heater
2nd floor: Test section - cooler
3rd floor: Separator
roof level: Aerocondensor

3.2.2. Temperatures and pressures are measured at preferential points of the loops, particularly at the inlet and outlet of the test sections. Data on the upstream conditions are obtained from the measurements and the enthalpy balance in the heater.

In general, the thermodynamic states of the fluid—one or two phase—between the pump outlet and the test section outlet are checked on a pressure-enthalpy diagram.

This diagram is employed to recalculate the pressure drop evolution in the test section by using the correlations derived from the hydrodynamic tests. In fact, the discrepancies between the calculated pressure drops and measurement results may in some cases amount to as much as 50%, particularly in the case of highly subcooled flows at the test section inlet.

The difficulty stems from the assumptions made in the zone in which net boiling predominates (the void fraction is already large, whereas the total quality is zero).

3.2.3. A burn-out detector has been developed with a view to preventing the systematic destruction of the heating tubes at each measuring point.

The detection element consists of a pyrex tube on which an 8/100 mm constantan wire is coiled at 2 windings per mm.

By coating half the circumference over the entire length of the coil, it is intended to make two copper-constantan thermocouples generatrixes.

One of the generatrixes is insulated by a coating of refractory cement, while the other (so called sensitive) is placed opposite the heating surface.

A fairly rapid temperature rise on the heating surface will act by radiation on the bare thermocouples without affecting the thermocouples provided with heat insulation. These latter thermocouples will act as cold welds for the whole system.

The large number of thermocouples in series makes it possible to detect very small temperature variations in relation to each thermocouple.

In the case of a protected circular test section, four detectors of this type are placed, diametrically opposed, around the test tube, with the sensitive generatrixes facing inwards.

In order to protect the heating elements of the four-rod section, the detectors are placed inside the power generating tubes.

The detectors are therefore fitted with four sensitive generatrixes and four heat-insulated generatrixes in order to detect a possible heating in azimuth (Figure 12).

The signal obtained makes it possible, by means of an electronic device, to cut off the electricity supply to the test section.

Voltage points are arranged every 10 cm along the coil of the detector: the signal induced in a section causes, at the same time as the feed circuit is cut off, that zone of the heating tube which is affected by the critical phenomenon to be indicated on another electronic device.

3.3. *Results*

For the details, reference will be made to the test results set out in EURAEC reports nos. 145, 146 and 147.

The tests carried out on the various geometries yield results which it is difficult to fit into a coherent pattern, taking into account the different hydrodynamic conditions encountered in the test sections and the absolute pressure levels at which the experiments were performed (in the case of the 10 mm diameter tube, for example, flux and flow rate variations give rise to appreciable variations of pressure at the critical point, which does not happen in the case of the four-rod test section).

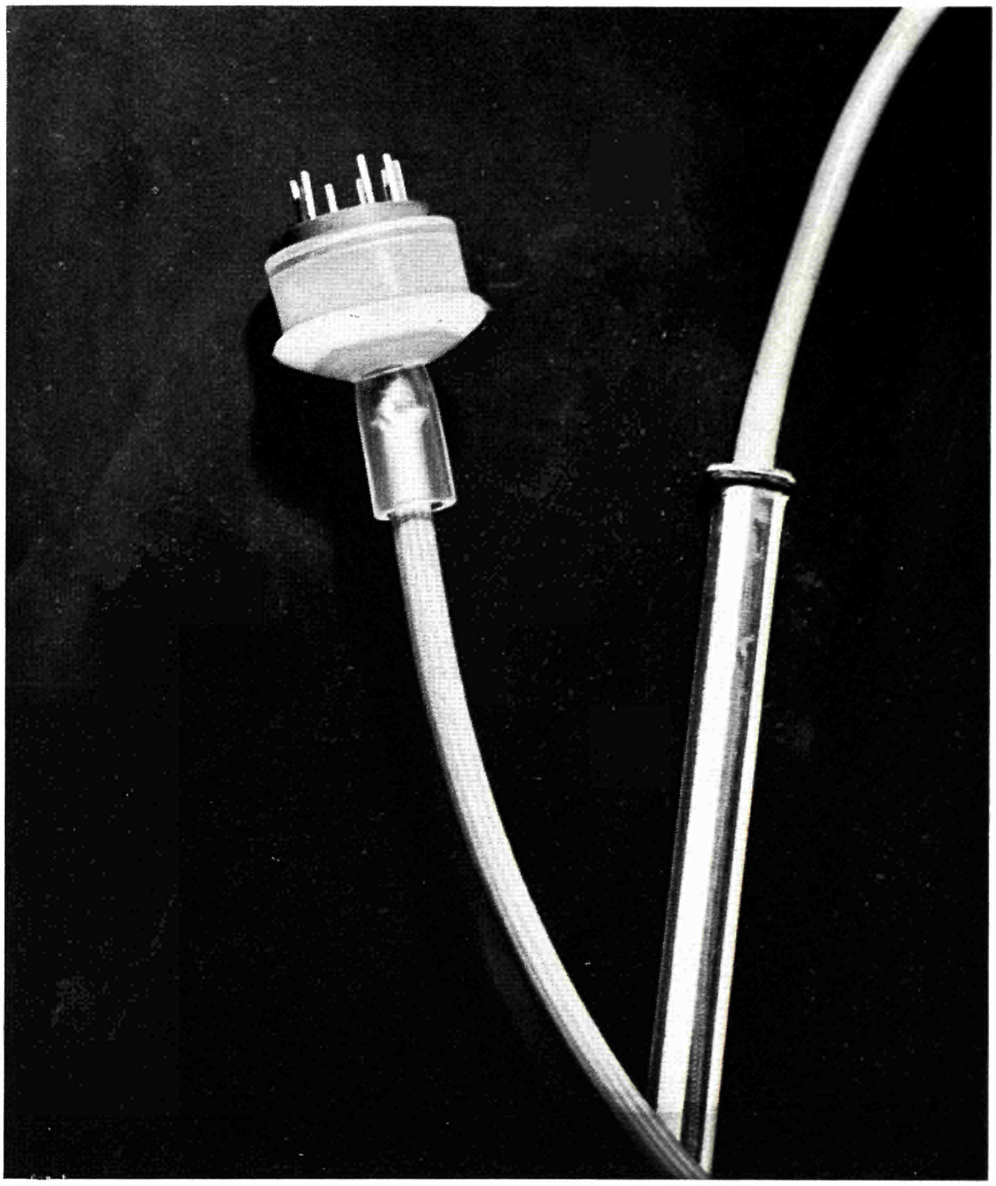


Figure 12
Detector with 4 sensitive generatrixes and six independant sections.

Two significant results, however, may be singled out:

1) shifting up to a higher level of the burn-out heat flux by the introduction of twisted tapes in the test section:

— at a given flux, burn-out is given at an appreciably lower flow rate in the case of vortex flow.

The ratio between the flow rate at straight flow and that at vortex flow amounts to 2 (quality doubled at test section outlet);

— at a given flow rate, the burn-out flux can be multiplied by 1.6. These results are illustrated by figures 13, 14 and 15.

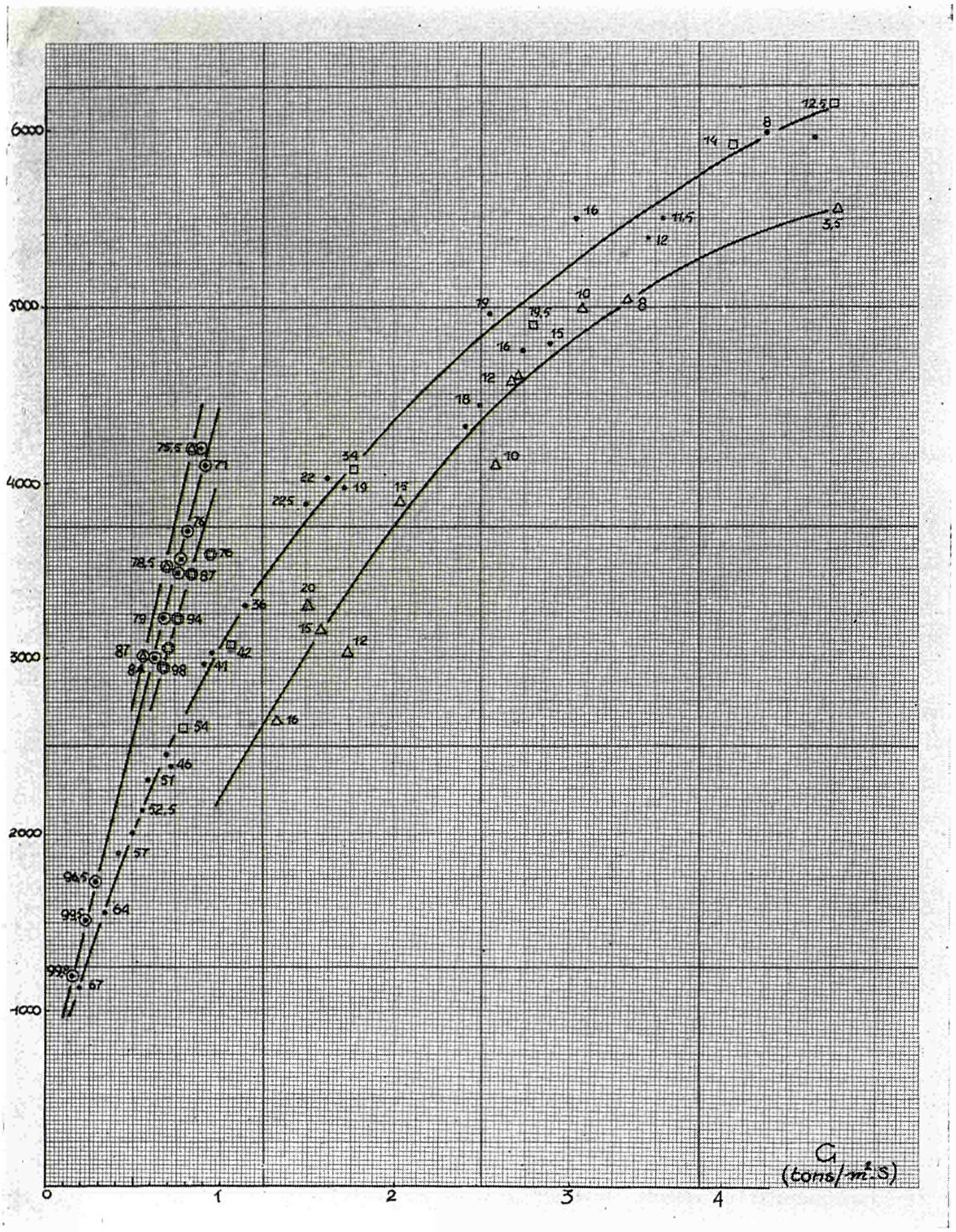


Figure 13

Crisis heat flux.

Test tubes ∅ 10 × 13 mm, length 800 mm.
△ Results without tapes and cooling before T.S.
□ Results without tapes and heating before T.S.
• Results without tapes with inlet T.S. enthalpy of 430 kJ/kg.
(△) (□) (•) Results with internal twisted tape $y = 3$, same conditions as before.
Figures in front of points represent the steam quality at the outlet of the T.S.

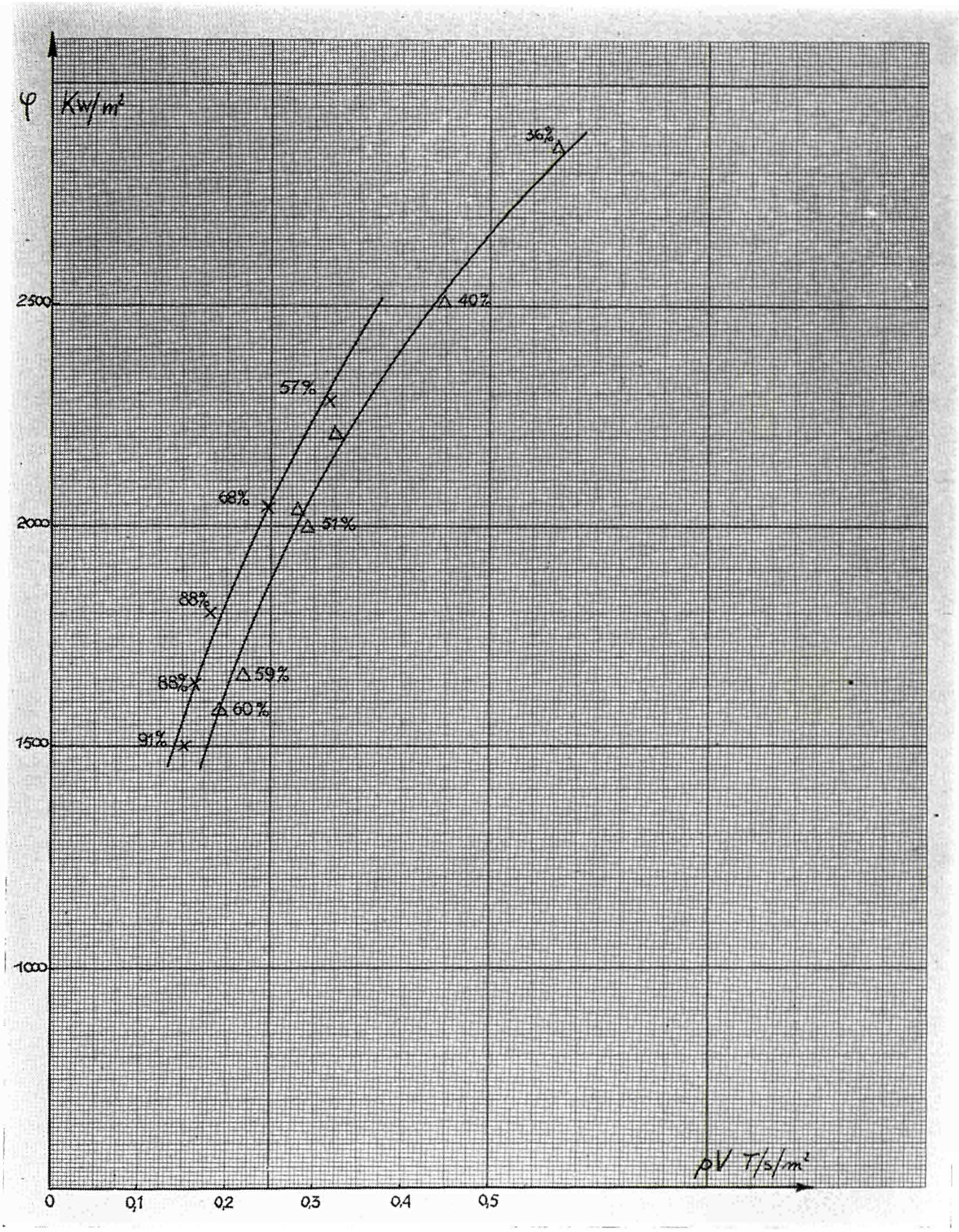


Figure 14

Crisis heat flux versus flow rate.

△ Tube ∅ 20 without twisted tape
× Tube ∅ 20 with twisted tape $y = 3$
heated length: 800 mm.

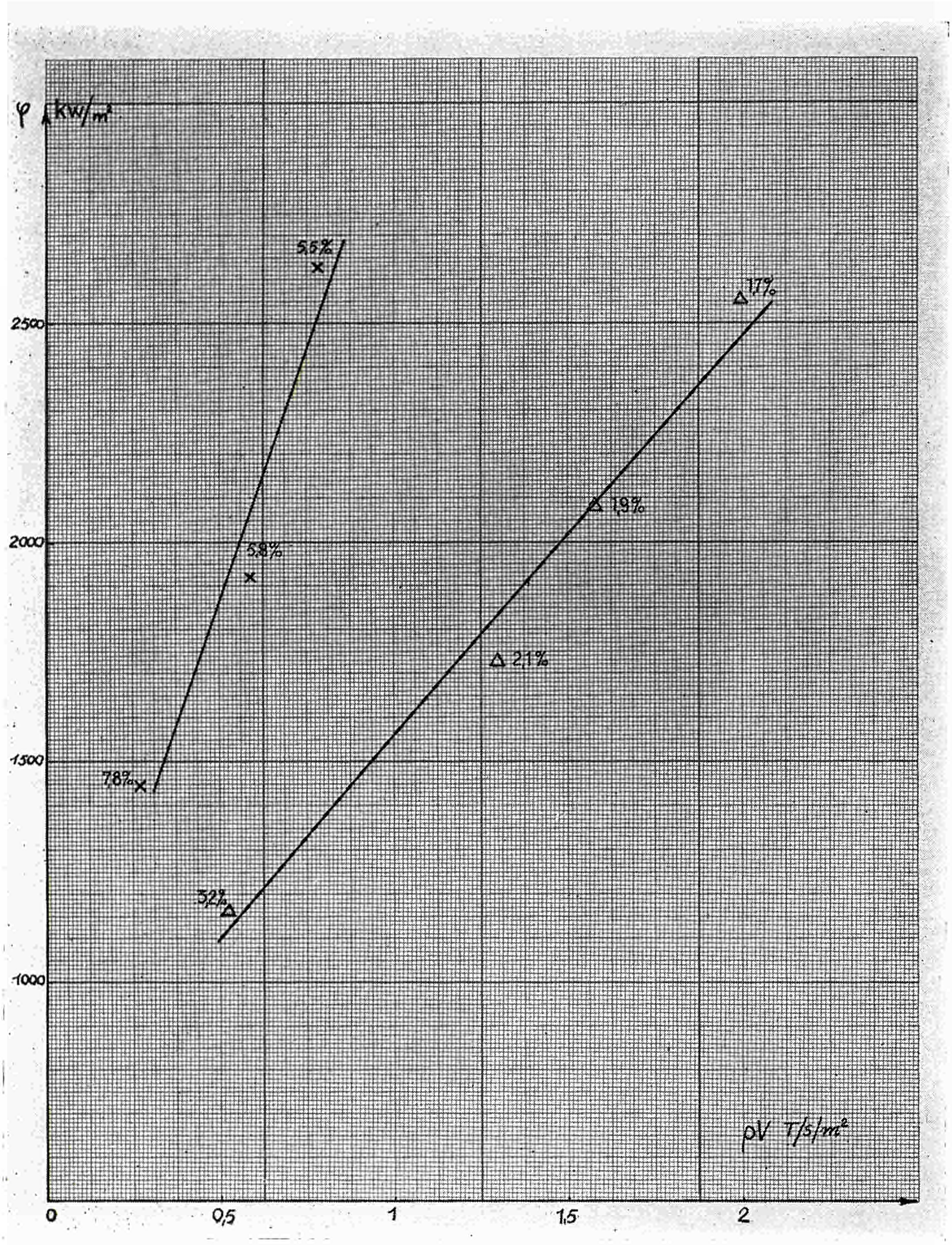


Figure 15

Crisis heat flux versus flow rate.

Annular section: heated rod: Ø ext: 11 mm
L = 600 mm
shroud: Ø int.: 31 mm
△ Without twisted tape
× With twisted tapes $y = 3$

2) the disappearance of unstable flows

— it is found by experiment that the use of spirals, although not eliminating them altogether, at least attenuates the flow vibrations obtained under conditions thermodynamically identical with straight flow, particularly at steam qualities of less than 10% at the test section outlet.

These phenomena may be explained by the disappearance of slug flow which is incompatible with new flow formation.

4. CONCLUSION

The results obtained at low pressure need to be checked by means of the experiments now in progress on the installation which can be used to reproduce the temperature and pressure conditions of a BWR.

A study is likewise under way with a view to defining the consequences, from the neutronic standpoint, stemming from the introduction of a material in the form of the twisted tapes between the rods of a fuel element.

At the same time, a technological study will make it possible to define, by means of a real element, the attachment and leaktightness fittings with which the twisted tapes are to be equipped.

SUMMARY OF A RESEARCH PROGRAM IN TWO-PHASE FLOW IN PROGRESS AT CISE

Pr. M. SILVESTRI
C.I.S.E.: Segrate (Milano), Italy

TABLE OF CONTENTS

CHAPTER I

WORK MADE AT CISE ON TWO-PHASE FLOW AND RELATED PROBLEMS IN CONNECTION WITH NUCLEAR REACTOR COOLING

1. SCOPE OF THE WORK PERFORMED AT CISE

Work in progress at CISE on steam-water mixtures for reactor cooling (fog cooling) is at present essentially a research program performed in fulfillment of two contracts with Euratom, both under the supervision of the US-Euratom Joint Board.

These contracts, which cover the period July 1961 - December 1963, deal with several problems concerning fog cooling, i.e. heat transfer, hydrodynamics of two-phase flow, corrosion and erosion of structural materials (Zircaloy-2 and stainless steel) by steam-water mixtures flowing both in the absence of radiation and in a radiation field.

The scope of the present program is actually an extension of the tasks accomplished under previous contracts started at the end of 1959 and terminated during the last year.

Basic data were obtained in that period concerning the heat transfer properties of steam-water mixtures, the hydrodynamic characteristics of two-phase flow and the corrosion behaviour of structural materials. Critical (burnout) heat fluxes, pressure drops and heat transfer coefficients were measured in a wide range of mass velocities, qualities and pressures (mostly 1,000 psi), using a loop in Piacenza with electrically heated test elements both tubular and annular. Length to diameter ratios of the heated elements were also varied in these experiments. In another loop, the adiabatic flow of argon-water mixtures was investigated, with particular respect to phase distribution over the cross-section of the duct, liquid film thickness at the wall of the duct, pressure drops and pressure oscillations. Mixing and separation of the two phases were studied in two loops (one with air-water and the other with steam-water at 800 psi) operated by Ansaldo in Genova. Corrosion data were obtained both in static and dynamic conditions. Tubular specimens were inserted in a loop, specially designed and constructed, operating at 1,000 psi.

In the extended program the main tasks are the following:

a) heat transfer experiments with the Piacenza loop in a broader range of length to diameter ratios; measurements of heat transfer coefficients with increased accuracy; determination of critical heat fluxes in transients and with tubes non-uniformly heated;

b) heat transfer experiments with the Ansaldo loop in Genova, duely modified, allowing larger duct cross-sections and thus more elaborate test element configurations;

c) hydrodynamic experiments in adiabatic conditions with the argon-water loop in order to investigate more carefully the influence of the physical properties of the fluids (such as density, viscosity and surface tension) on the flow distribution of the two phases; measurement of the flow rate of the liquid film flowing at the duct wall; shadow effects of spacers and other obstacles; flow stability in channels connected in parallel;

d) corrosion experiments with controlled oxygen on hydrogen content in the steam-water mixture; effects of surface finish on corrosion rate; erosion in case of significant impact angles; corrosion in conditions of "slow burnout";

e) in-pile experiments to check the corrosion data under radiation and to study the associated chemical problems such as radiolysis, coolant activation and contamination by activated corrosion products.

In order to have a more complete picture of the potential interest of steam-water mixture for reactor cooling, problems in the fields of reactor physics, engineering and preliminary design were also taken into consideration, although they are not included in the above mentioned contracts.

While details of this work are given in paragraph 4, it has to be pointed out here that particular emphasis was put on the fog cooled heavy water moderated concept—especially in reactor physics experiments because, according to CISE's philosophy, it seems to offer a greater flexibility in design than other concepts (e.g. graphite moderated), which also appear feasible

2. HEAT TRANSFER AND HYDRODYNAMICS

Heat transfer experiments for the measurement of critical heat fluxes, heat transfer coefficients and pressure drops are being performed in a loop (installed in the Piacenza Power Station), the scheme of which is given in Figure 1.

The loop includes an electrically heated (800 kVA max) once through boiler capable of supplying steam at the desired flow rate and inlet quality to the test section. The test elements, also electrically heated (max. power about 100 kW D.C.), consist of round tubes and round annuli heated internally, externally or at both walls. The critical heat flux is determined as a function of quality, mass velocity, pressure, length-to-diameter ratio and coolant channel geometry. Along with the heat flux measurements, the pressure drops across the test section are measured with and without heating. Also, the heat transfer coefficient below and (in some cases) above critical conditions is evaluated from the measured data.

The experiments were performed in the following ranges of the principal parameters:

— length to diameter ratio: 4 - 300

— pressure: 40 - 80 atm

— mass velocity: 100 - 400 g/cm^{2}sec

— steam quality: -0.1 to ~ 1

Round tube diameters are in the range between 5 and 10 mm, the upper value being determined by the maximum total flow rate of the loop (about 1,100 kg/h).

Heat transfer experiments in channels of larger cross-section will be shortly initiated in a loop installed at the Ansaldo Mechanical Works in Genova; the scheme of this loop, which operates at about 55 atm, is shown in Figure 2. Superheated steam at 350°C and 60 atm is produced in an oil-fired boiler, with a maximum flow rate of 10 t/h. After steam de-superheating and mixing with water, the resulting mixture passes through the test section. At the top of the test section steam is separated in an especially developed device and then condensed.

Electrical heating of the test elements is provided by a set of 28 D.C. generators, connected as shown in Figure 3, totalling about 1,200 kW maximum power.

In the Genova loop several test element configurations will be investigated, including tubular and cluster geometries. Experiments are foreseen to study some hydrodynamic aspects of the two-phase flow in the channels, like the phase distribution in the cross-section.

For the measurement of hydrodynamic quantities several techniques will be applied, which were previously developed in separate hydrodynamics studies. These were performed—and are still pursued—in a special loop (installed at CISE), in which a mixture of argon and water at room temperature and at a pressure of about 22 atm flows through the test section in adiabatic conditions.

The scheme of this argon-water loop is illustrated in Figure 4. Experiments are performed in the same range of mass velocities and qualities considered in the heat transfer program, while the operating pressure was selected so as to have a gas density equal to that of saturated steam at 70 atm.

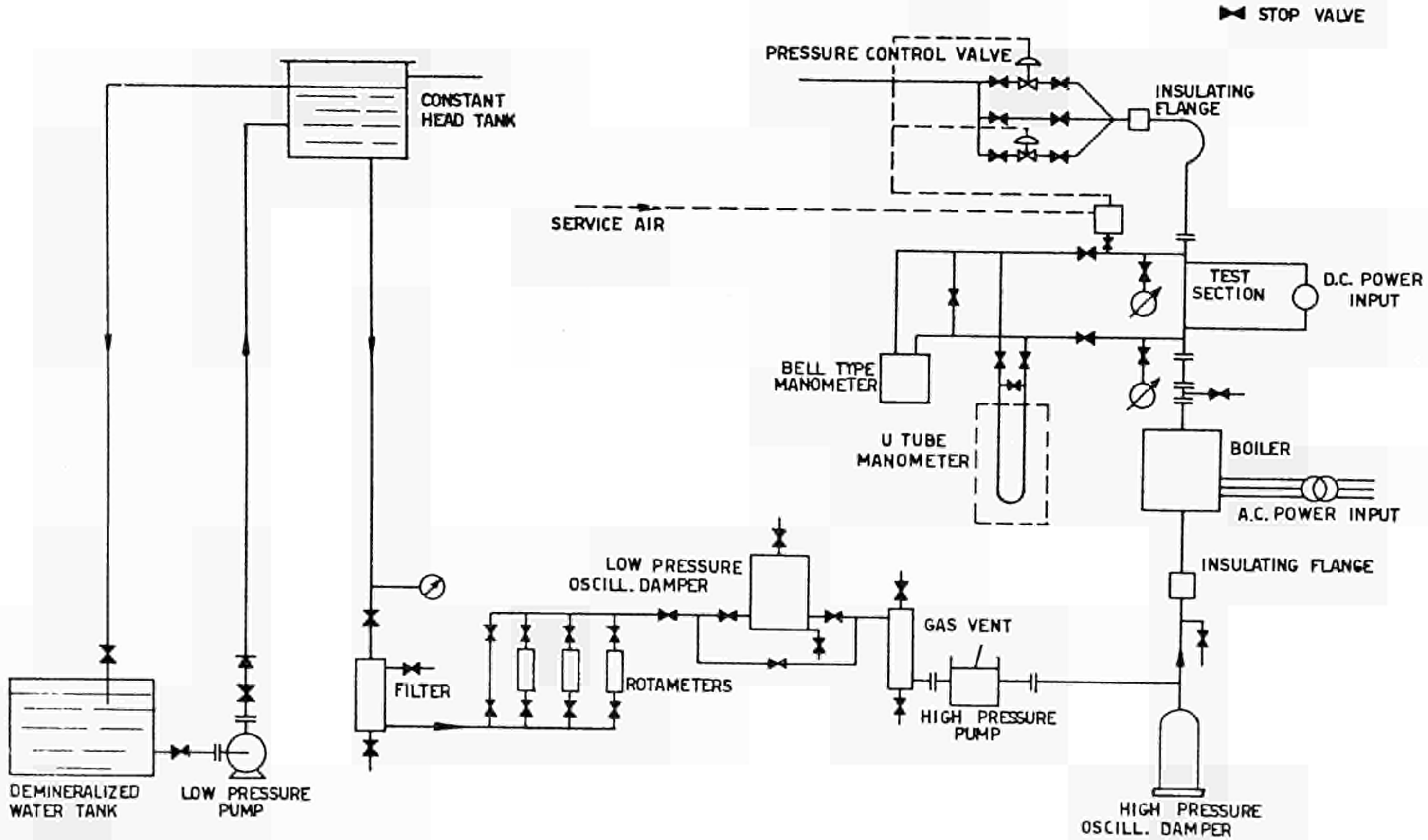


Figure 1

Heat transfer loop operating in Piacenza.

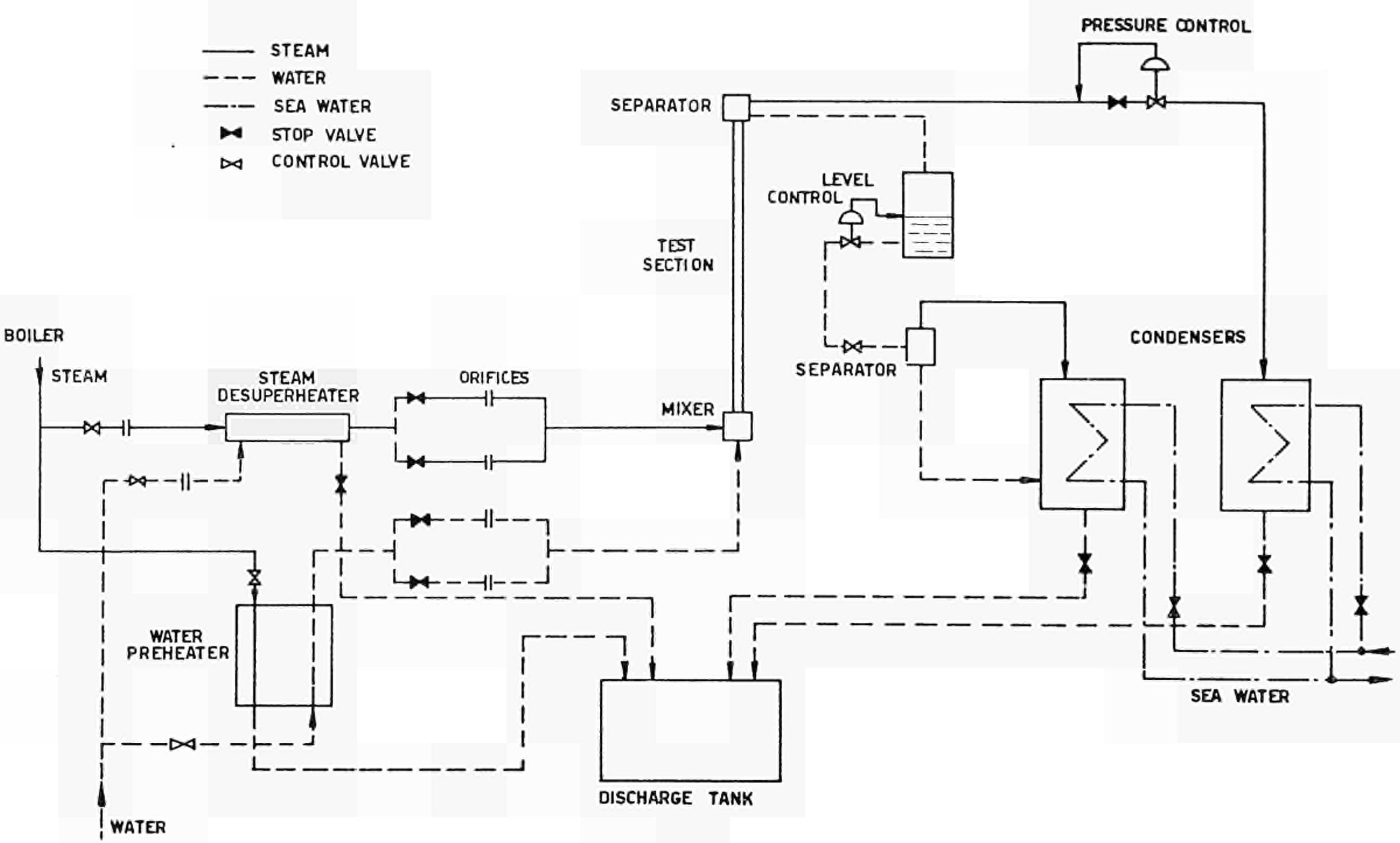


Figure 2

Heat transfer loop in Genova: flow sheet.

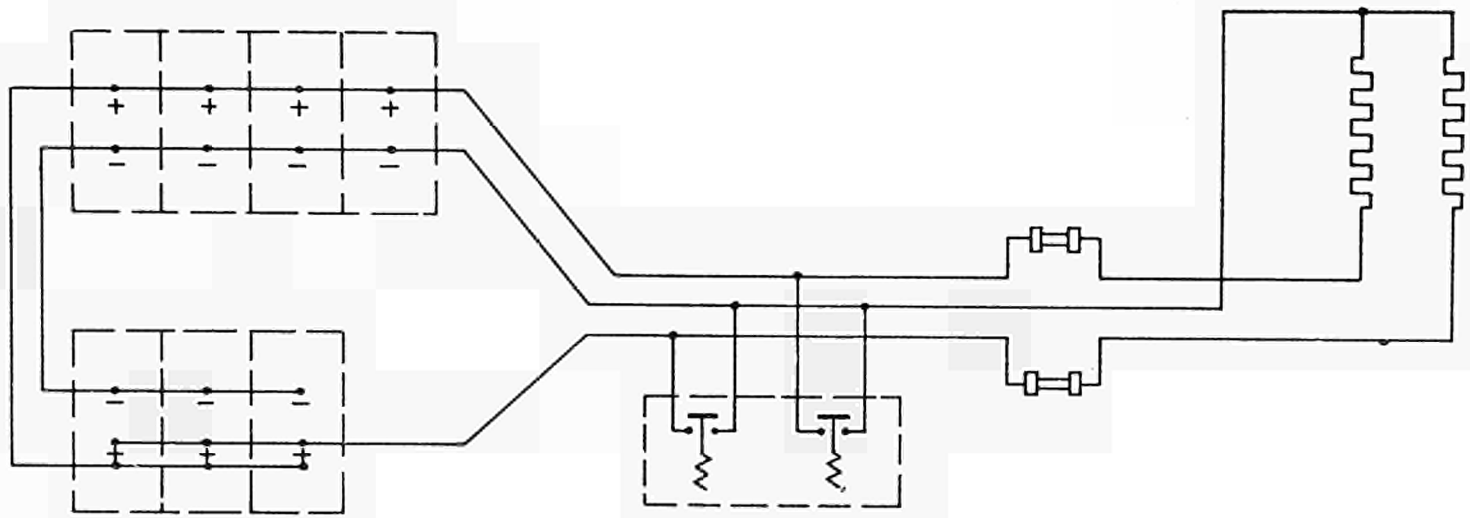


Figure 3

Heat transfer loop in Genova: D.C. power system.

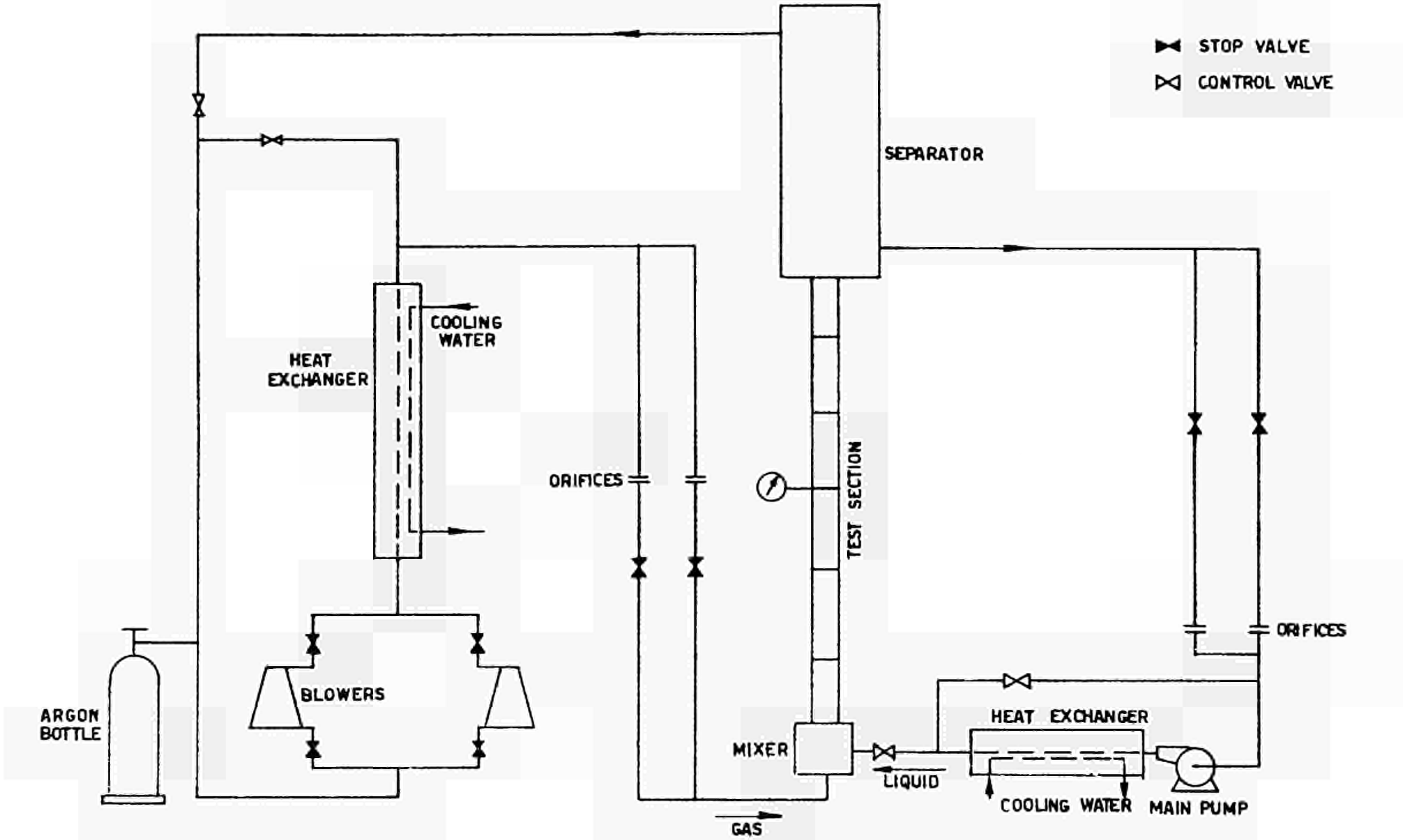


Figure 4

Loop for hydrodynamic experiments.

The configuration of suitable devices for mixing and separating the two phases was previously studied at Ansaldo Mechanical Works in an air-water loop at atmospheric pressure. The screening of these devices led in particular to the development of a steam separator (see Figure 5) highly efficient and compact. Further tests performed in the Genova steam facility have shown a separation efficiency higher than 99% with mass velocity of 300 g/cm^{2}sec and 40% quality. For a total steam-water mass flow rate of 15 t/h (about 33,000 lb/hr) dimensions of the separator are: 43 cm height and 9.5 cm O.D.

3. CORROSION AND IN-PILE EXPERIMENTS

Corrosion and erosion of Zircaloy-2 and stainless steel by steam-water mixtures are being studied in two almost identical loops. One of these is illustrated in Figure 6. The steam-water mixture is produced at the desired quality in a boiler, consisting of an electrically heated coil of tubing. After passing through a set of tubular corrosion specimens connected in series (see Figure 7), the mixture is cooled, and the resulting water is depressurized and pumped again to the boiler.

The behaviour of the materials is examined at several exposure times in the following ranges of the involved quantities:

temperature	from 245 to 300 °C
mass velocity	from 270 to 550 g/cm^{2}sec
steam quality	from 30 to 65%.

The corrosion rate of Zircaloy-2 in flowing wet steam appears to be always higher than in static conditions. No evidence of erosion was found in straight tubes even at the highest mass velocities (corresponding to linear velocities up to about 150 m/sec), while some erosion effects were observed in bends. No influence was found on corrosion by heating the specimens by D.C. current.

Corrosion is studied by weighing the specimens before and after exposure. Specimens are also subjected to metallographic examinations in order to observe the oxide layer and to investigate the hydride formation due to hydrogen absorption.

The influence of a controlled content of oxygen or hydrogen in the steam-water mixture is also being investigated. These experiments are performed in a loop, which differs from that described in Figure 6 only by the fact that the polythene tank at atmospheric pressure is substitued by a pressurized tank for dissolving oxygen or hydrogen into water.

Zircaloy-2 corrosion experiments will be also performed in an in-pile loop without fuel, which will be placed in the 2 MW swimming pool reactor Avogadro-1 (Sorin, Saluggia). This loop, which will soon start operation, is illustrated in Figure 8. It is quite similar to the other two corrosion loops, except for some peculiar features mostly concerning the safety aspects and for the test section, which was specially conceived for operation close to the reactor core.

Tests will be performed in the same ranges of the parameters considered in out-of-pile experiments.

In addition to the long exposure tests for investigating the influence of radiation on corrosion, experiments are also foreseen in order to study the entrainment of activated

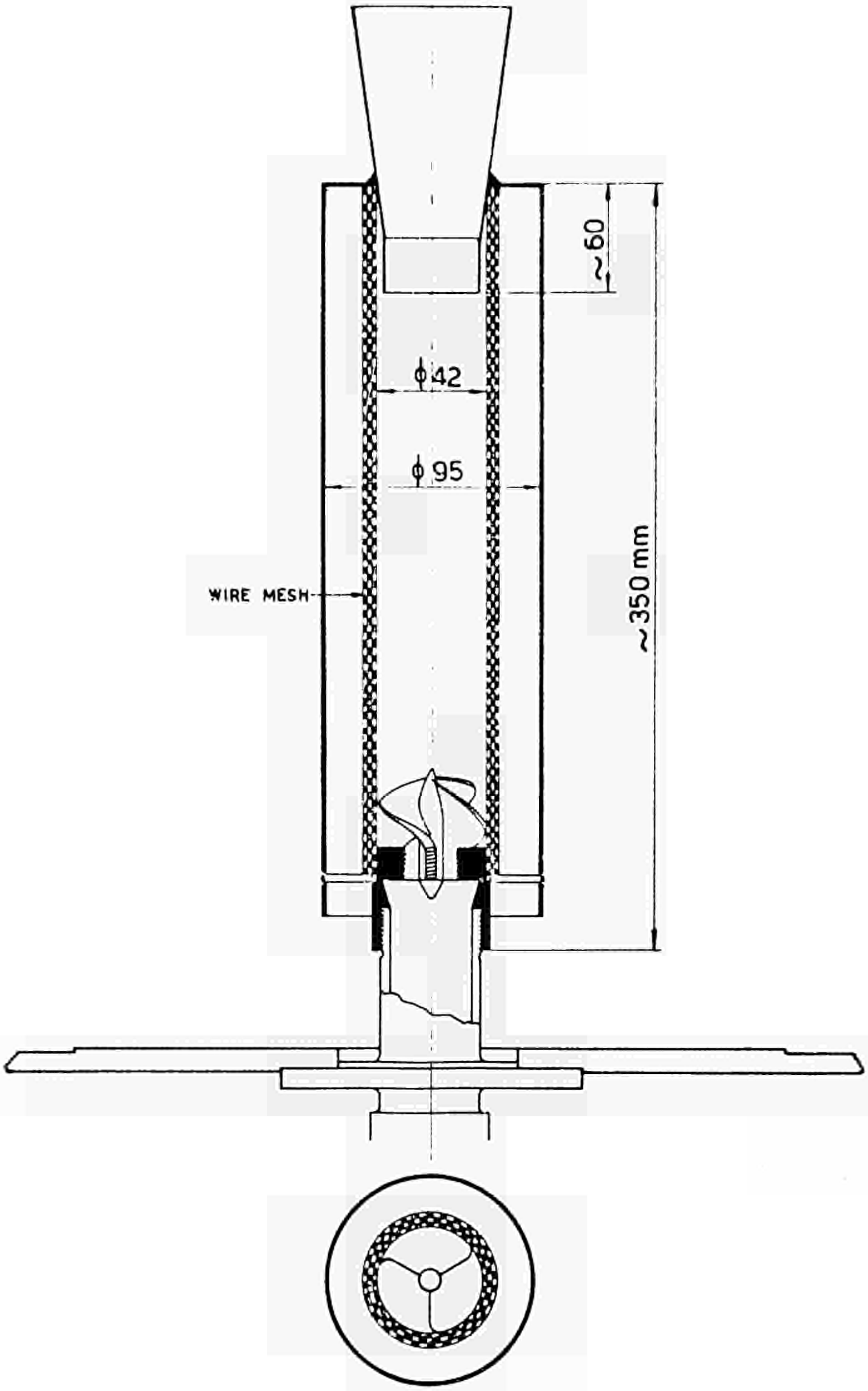


Figure 5

Steam-water separator developed at Ansaldo.

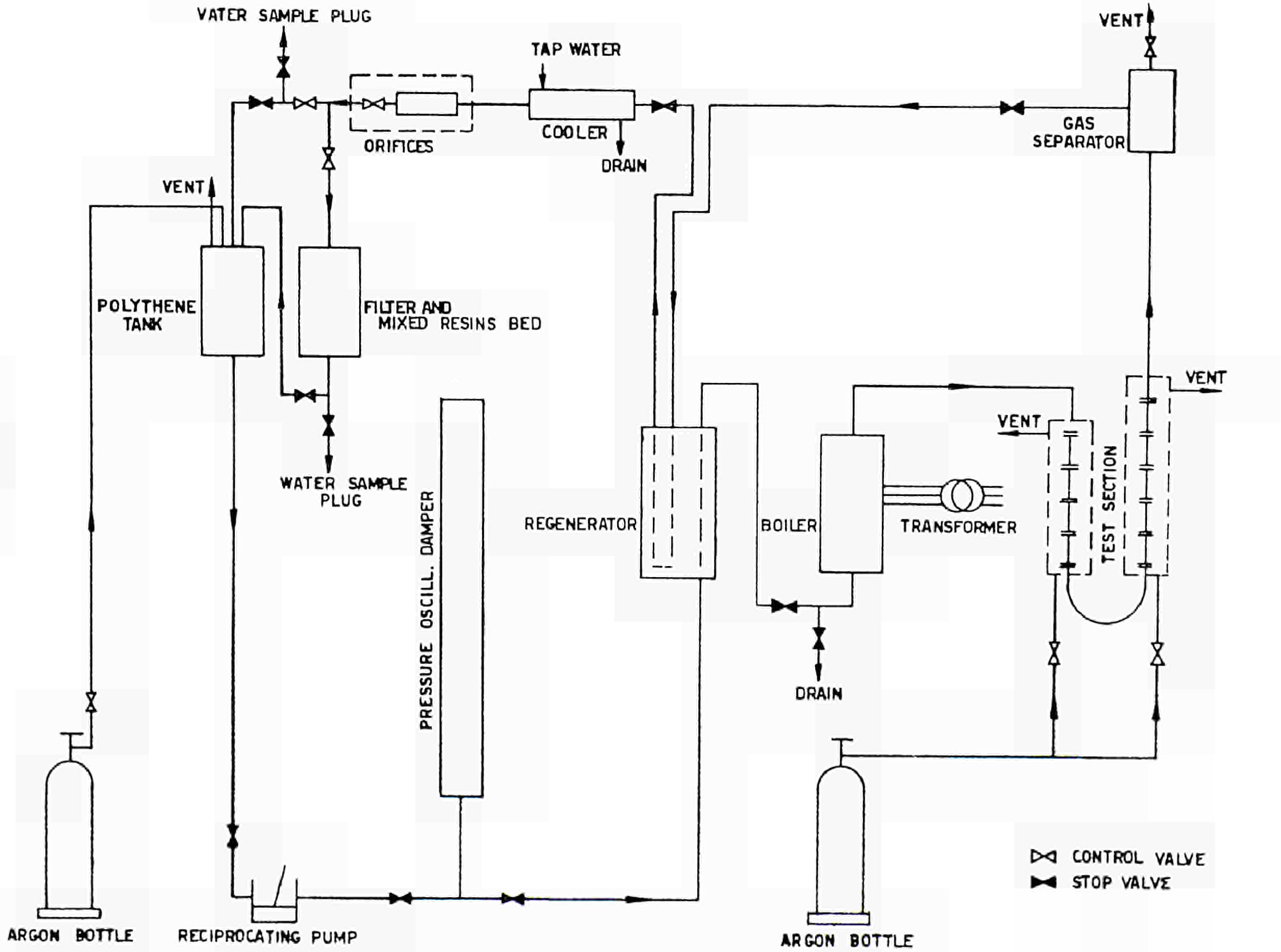


Figure 6
Corrosion loop.

Figure 7

Tubular specimens for corrosion experiments.
Top: Exploded and assembled view of a joint.
Bottom: The same joint with power lugs for heating the specimen.

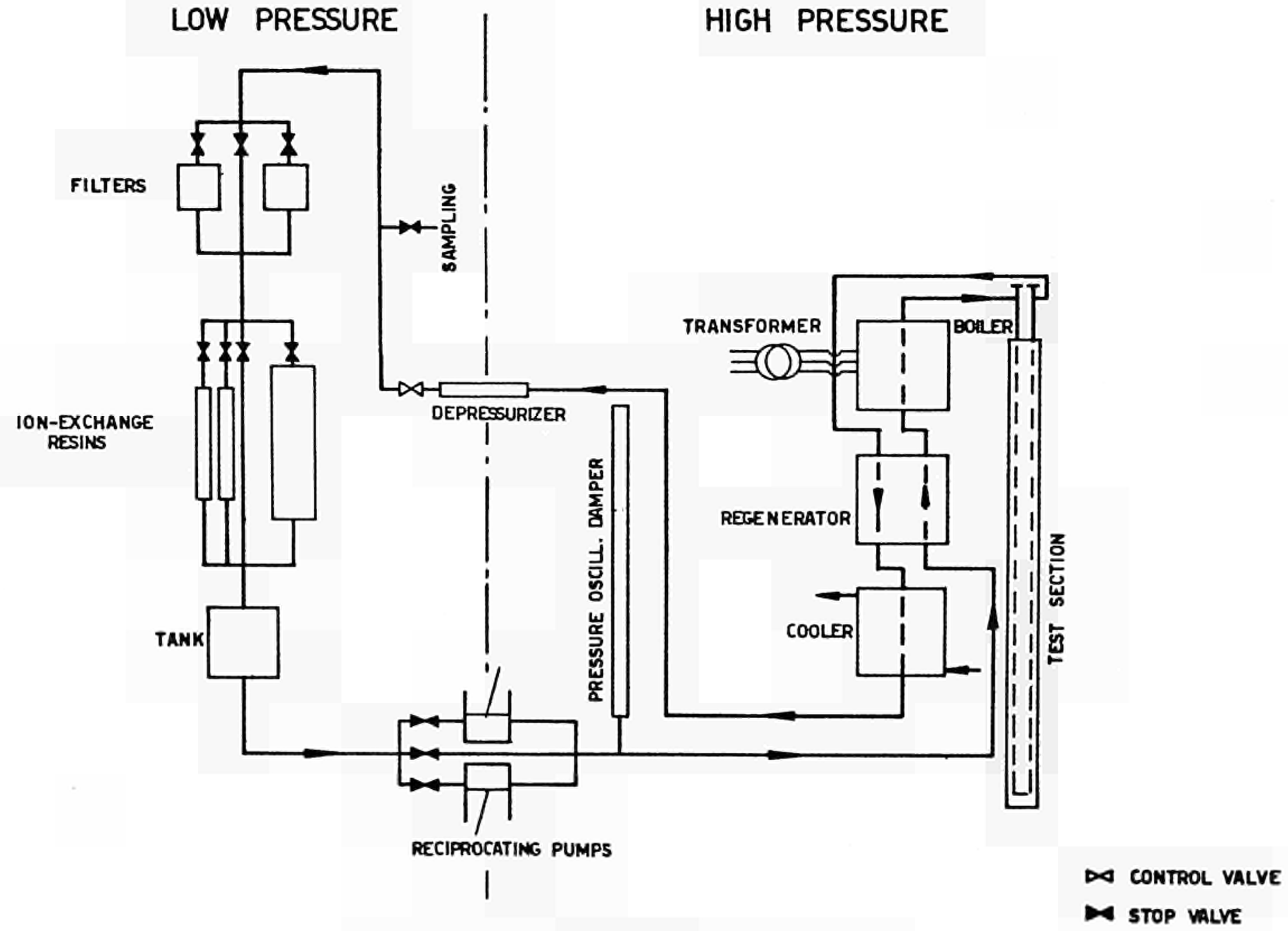


Figure 8
In-pile loop for corrosion and water chemistry experiments.

crud by the mixture, the purification of water from activated dissolved substances and the formation of radiolytic gas. Suitable analytical methods were developed for this purpose.

4. REACTOR PHYSICS, ENGINEERING AND DESIGN

Reactor physics experiments and calculations are mainly devoted to the study of the heavy water moderated concept. A digital computer code for reactor criticality calculations was prepared and is now used in an Olivetti machine. Two additional codes are now being developed, one for long term reactivity changes and another for reactor dynamic studies. An important problem, i.e. neutron thermalization in a lattice having two moderating media at different temperatures, has also been studied.

Criticality experiments with different configurations of heavy water moderated lattices were performed on Aquilon-2 critical assembly at Saclay, by the replacement method. Fuel elements were made up with concentric annuli of natural uranium metal and low density polystirene, simulating the steam-water mixture. A picture of one such element is shown in Figure 9.

Another set of experiments will start shortly on the RB reactor (Vincha - Beograd) in order to investigate the flux flattening in a heavy water lattice having an inner reflector containing a cylindrical black surface.

Experiments on neutron thermalization in a water moderating medium at two different temperatures are now in progress by means of the pulsed neutron technique. The flow sheet of the loop used for this purpose is shown in Figure 10.

On the basis of the experimental results obtained at CISE, preliminary studies have been recently undertaken to investigate to a first approximation the main problems involved in the design of a power reactor cooled by a steam-water mixture.

The first problem has been the interpretation and the utilization of the critical heat flux data. The criterion for a safe operation was selected by taking the crisis as a local phenomenon and limiting the local value of the steam quality in the most dangerous section of the conduit. The question will be reviewed by considering the total power input as well as the possibility of operation above the critical heat flux with high quality mixtures.

Another major problem is the design of a suitable fuel element for a pressure tube fog cooled reactor. The tubular geometry, with internal cooling (Figure 11a), was selected and calculations have been made for metallic fuel (uranium or alloys) cladded by Zircaloy with a metallurgical bonding. The heat transfer between the fuel and moderator, both in steady and emergency conditions, was also investigated.

To study, in a very preliminary way, the feasibility of a fog cooled graphite reactor, the Sizewell reactor was taken as a reference design in which the actual coolant and the fuel elements were replaced by steam-water mixtures and single tubular elements (1.7 cm I.D.). With saturated water at the inlet, the resulting mean density was about 0.25 g/cm^3. The available reactivity was just slightly lower than in the gas cooled reactor (but with a higher ICR), although no optimization of the system was performed. The calculation made for the fog-graphite reactor will be used soon to investigate the control problems.

At present a cooling scheme such as that represented in Figure 11b, with flashing of water preheated in some channels inside the reactor, is being investigated for comparison with the case of subcooled water at the inlet of all the channels.

Figure 9

Top of a fuel element used for the critical experiments in the assembly Aquilon-2 at Saclay.

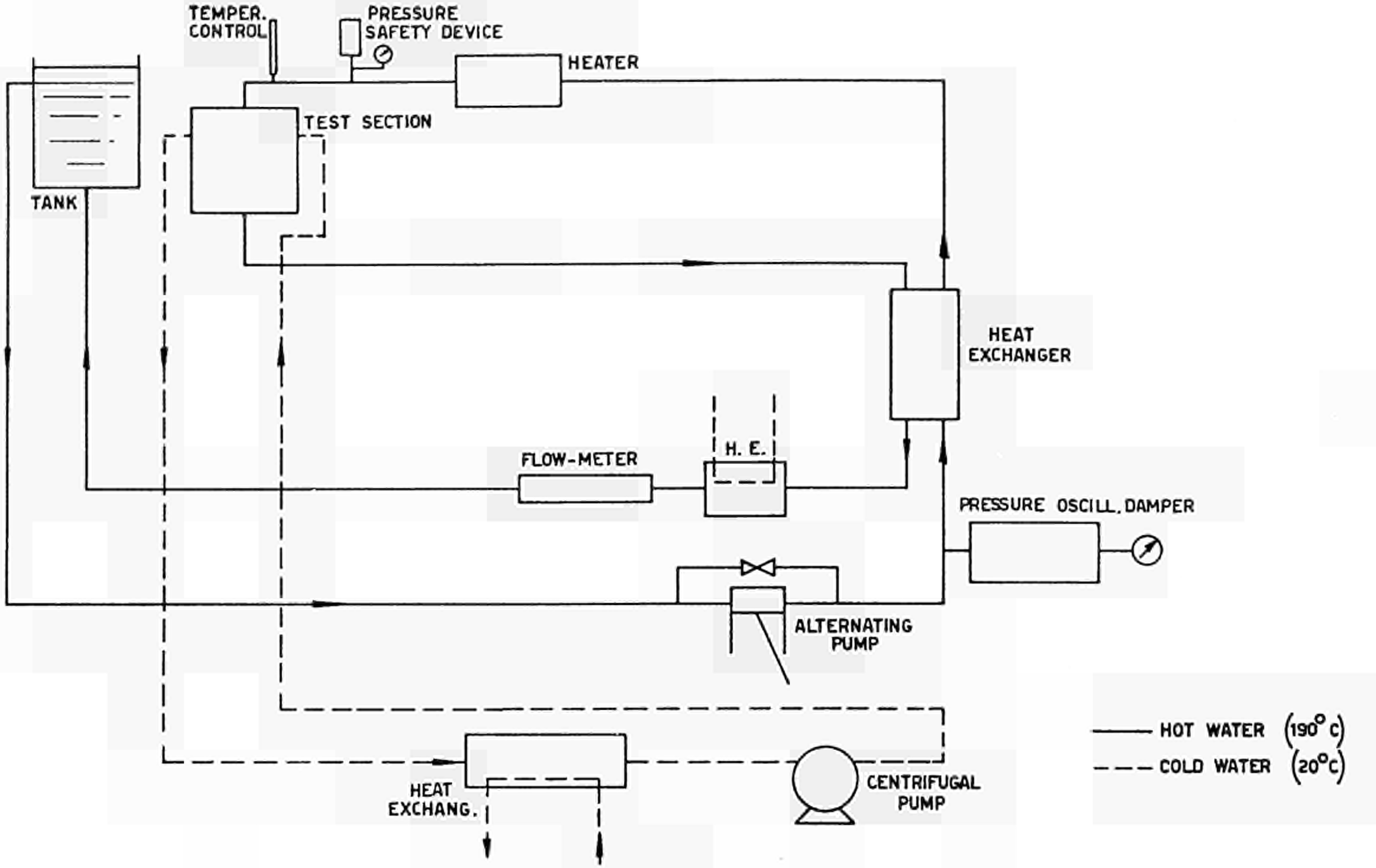


Figure 10

Loop for neutron thermalization experiments in a dual temperature water moderating medium.

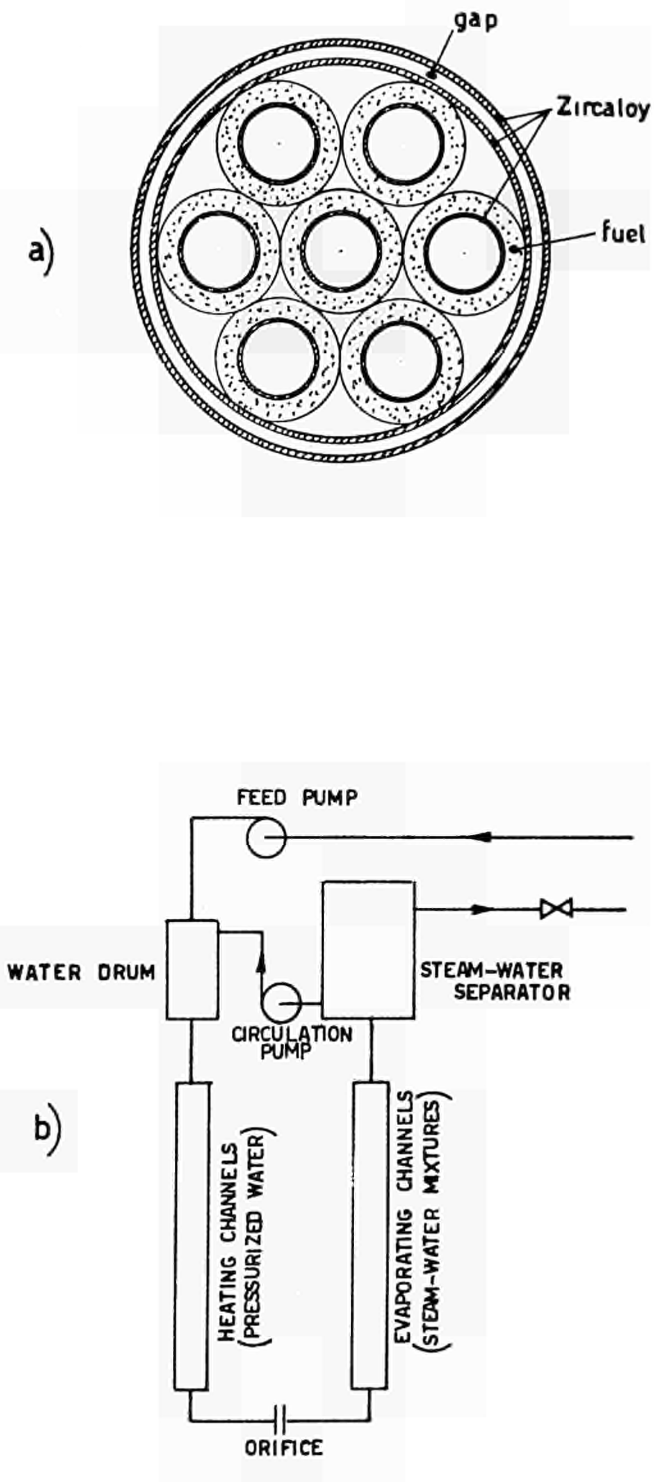


Figure 11

Fuel assembly cross-section (a) and scheme of a possible cooling circuit (b).

CHAPTER II

PRESENTATION OF SOME EXPERIMENTAL RESULTS OBTAINED AT CISE ON TWO-PHASE FLOW HYDRODYNAMICS AND HEAT TRANSFER

1. HEAT TRANSFER

In accordance with the provisions of the CAN-1 and CAN-2 program, sponsored by U.S.-Euratom joint Research and Development Board, critical heat fluxes, heat transfer coefficients, pressure drops with and without power in steam-water mixtures flowing upwards in round vertical tubes have been measured. Almost all the experiments were performed in fully developed flow, which was obtained through use of a calming section (2.5 m) and an exit section (3.8 m) of the same diameter as the test element. Moreover a number of experiments were carried out to test the influence of the variation of the inlet conditions (different calming sections, orifices).

All elements tested from November 1961 through October 1962 (CAN-2 program) are listed on Table 1. On the same table the geometrical dimensions, the flow rates and the type of measurements performed, are reported (in column type, Δp_0 means pressure drop without power, Δp pressure drop with power, ϕ_{B0} critical heat flux, h heat transfer coefficient).

Details on this experimental program can be found on CISE R 62, R 63, R 69 and in later reports.

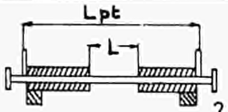
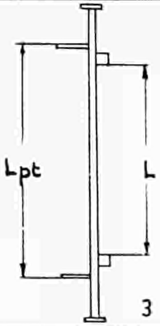
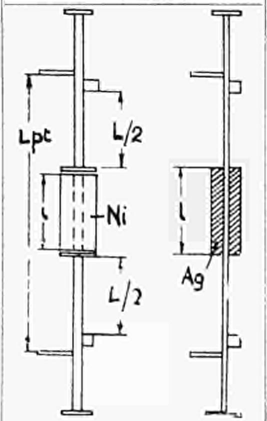
The critical heat fluxes versus X_{in} and X_0 for the lowest and the highest values of L/D are reported on Figures 2, 3, 4. From these plots it can be seen as the length (or L/D) has an adverse effect on ϕ_{cr}. This trend shows that the crisis is not merely a local phenomenon, but it can be visualized as an integral phenomenon along the whole test section. In fact the total critical power (Figures 5, 6) is less sensitive to L/D than the critical heat flux (a factor 30 in L/D corresponds to a maximum change in W_{cr} of about 3, while the change is of a factor of 10 for ϕ_{cr}). Moreover the W_{cr} becomes practically independent of length for L/D higher than $160 \div 200$.

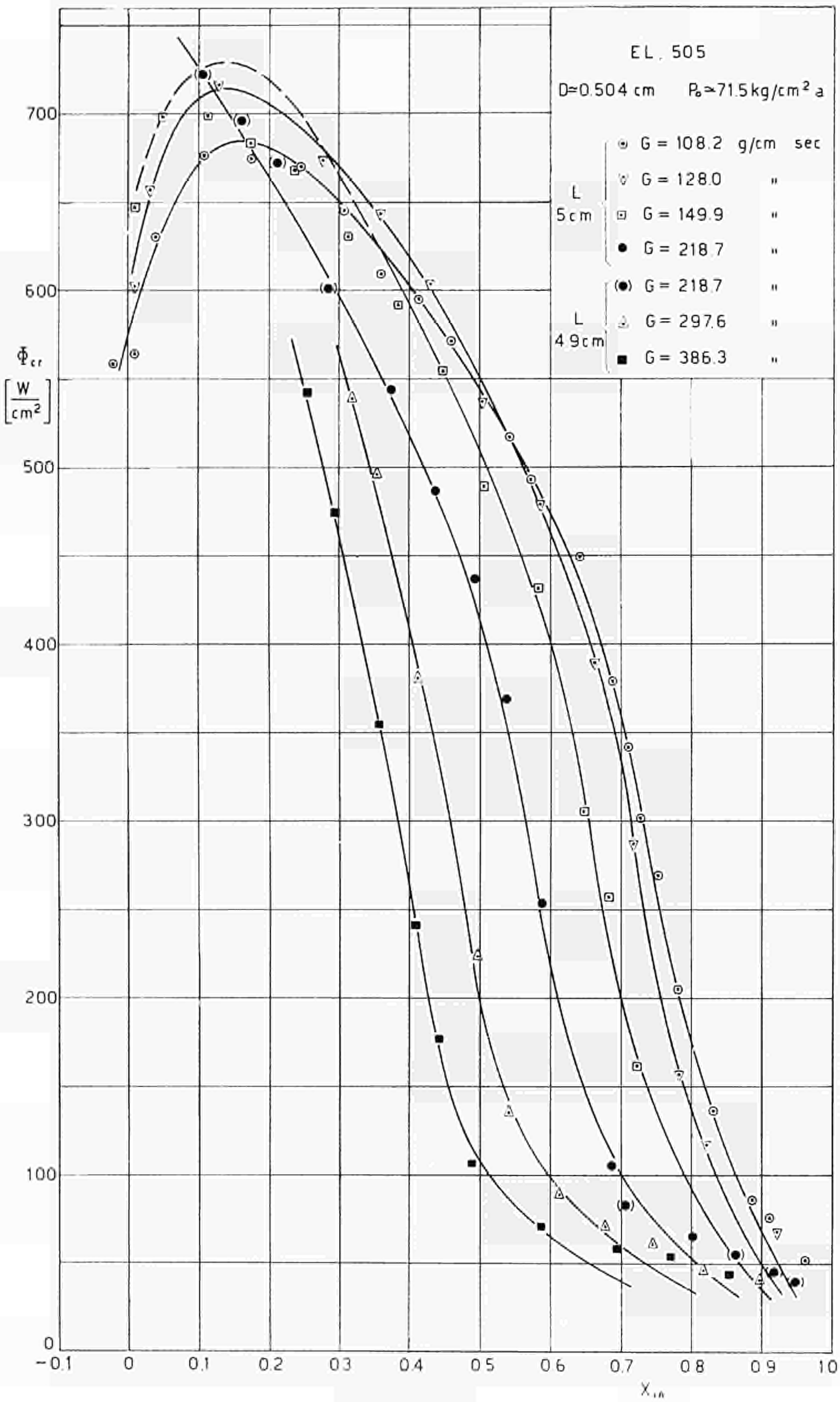
The ϕ_{cr} vs. X_0 curves (Figures 4) show how ϕ_{cr} can be, in a given range of qualities, a multi-valued functions of X_0. This characteristic disappears when X_{in} is selected as the independent variable (in place of X_0). This behaviour does justify the use of X_{in} instead of X_0 as an independent parameter the more that it is an input parameter, fixed a priori. Moreover the use of a local parameter as X_0 is no longer necessary if the crisis is interpreted as an integral phenomenon.

The choice of X_{in} as a main parameter is emphasized by Figure 7. The critical heat fluxes, obtained with an annular tube on the external surface with different heating on the internal rod, fall on the same curve when plotted versus X_{in} (at the same mass velocity); the same points vs. X_0 give different curves each one for a different internal power level, because X_0 is obtained by an enthalpy balance involving both external and internal power.

Further experiments to understand the L/D effect on the critical heat flux were carried out with elements having an intermediate non heated portion between two identical heated sections. A comparison between four test elements having the same

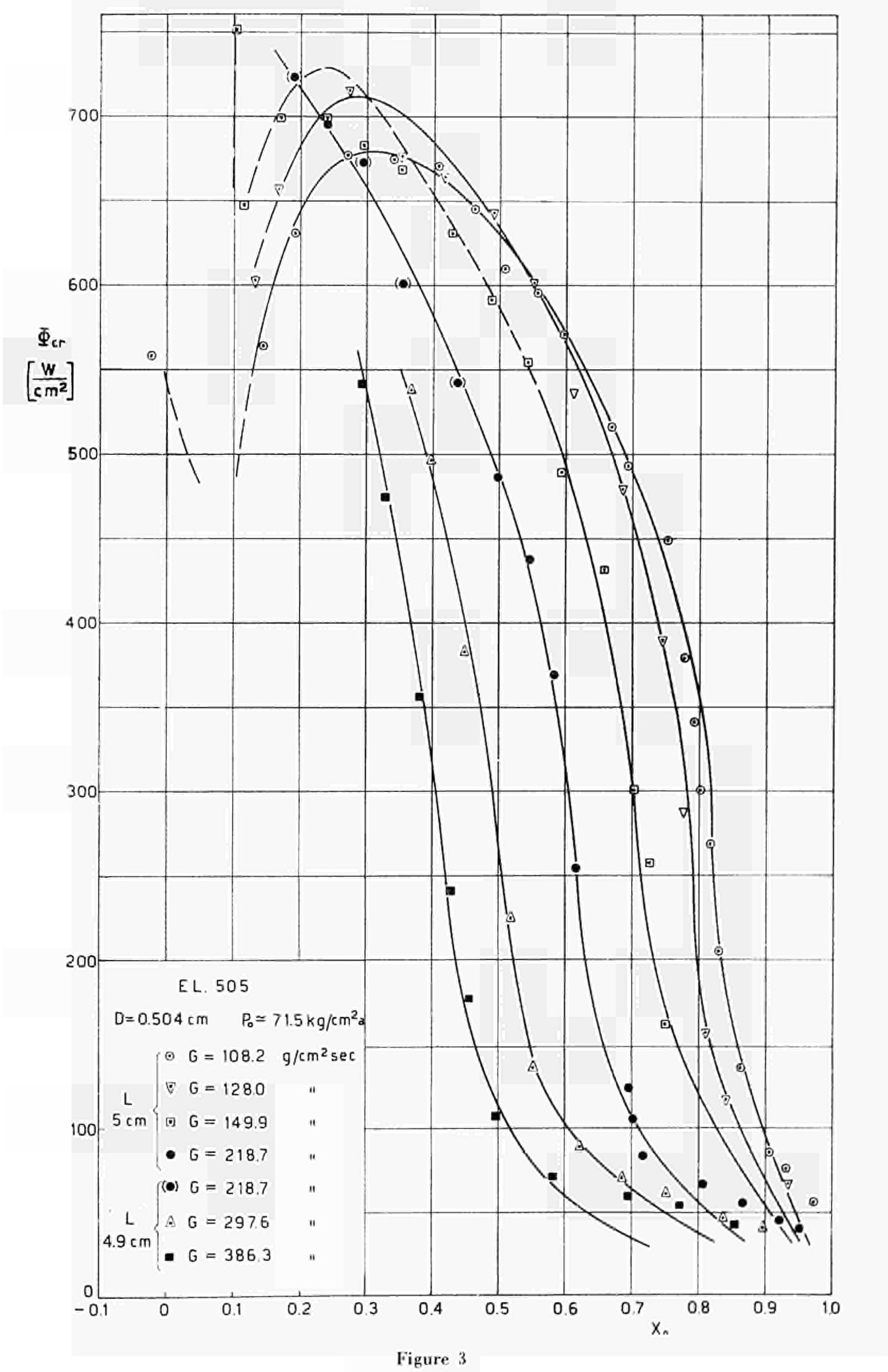
Table 1 — ELEMENTS TESTED FOR THE CAN-2 PROGRAM

	scheme	geometry	element	D inlet diameter (cm)	L heated length (cm)	$\frac{L}{D}$	L_{pt} distance between pressure taps (cm)	l adiabatic length (cm)	measurements type	specific mass flow rates g/(cm²sec)						pressure kg/cm²	notes
ROUND TUBES		1	500	0.508	—	—	102	—	Δp_0	106.6	125.3	147.6	215.7	291.5	380.3		For this element the $G = 175.5$ g/cm²sec was also tested.
	 2	2	502	0.504	2	3.61	11.2	—	$\phi_{Ho}-\Delta p$	108.6	127.9	150.1	219.7	297.5	386.8		Pressure drops with power are not valid because of the big difference between L and L_{pt}.
		2	505	0.504	5/4.9	9.92/1028	14.2	—	$\phi_{Ho}-\Delta p$	108.2	128.0	149.9	218.7	297.6	386.3		The heated length was reduced because holing of element.
	 3	3	51B	0.504	10	19.85	13.2	—	$\phi_{Ho}-\Delta p_0-\Delta p-h$	108.8	—	150.7	—	—	—		
		3	52B	0.508	20	39.4	22.0	—	$\phi_{Ho}-\Delta p_0-\Delta p-h$	106.5	125.6	147.5	215.6	292.6	379.8		It is the 560 element where only the second piece was heated.
		3	54B	0.500	40	80.0	42.0	—	$\phi_{Ho}-\Delta p$	110.0	129.7	152.6	222.4	302.2	392.2		
		3	58B	0.500	80	160.0	83.2	—	$\Delta T_{met}-\phi_{Ho}-\Delta p_0-\Delta p-h$	110.7	136.2	149.5	228.0	287.7	392.6		Critical heat flux presents some discrepancies when compared with other element.
		3	511	0.506	110	217.5	112.0	—	$\phi_{Ho}-\Delta p$	106.5	126.5	148.7	217.5	294.0	382.9	~71.5	
		3	515	0.508	150	295.1	152.0	—	$\phi_{Ho}-\Delta p$	106.3	125.5	148.0	215.6	292.0	373.8		
		4	521	0.508	40.1	—	43.1	1.0	$\phi_{Ho}-\Delta p-\Delta p_0-h$	106.5	125.9	147.7	216.7	292.1	380.2		
		4	522	0.508	40.1	—	44.1	2.0	$\phi_{Ho}-\Delta p-\Delta p_0-h$	106.3	125.7	147.5	215.7	292.6	380.3		
		4	523	0.508	40.1	—	45.0	2.9	$\phi_{Ho}-\Delta p-\Delta p_0-h$	106.3	125.8	147.9	215.8	291.8	379.9		
		4	525	0.508	40.1	—	47.17	5.17	$\phi_{Ho}-\Delta p-\Delta p_0-h$	106.2	125.9	147.8	216.5	292.2	380.4		
		4	530	0.508	40.1	—	52.27	10.17	$\phi_{Ho}-\Delta p-\Delta p_0-h$	106.5	125.8	148.0	216.2	291.9	380.1		
		4	540	0.508	40.1	—	62.15	20.05	$\phi_{Ho}-\Delta p-\Delta p_0-h$	106.2	125.6	148.1	216.6	292.1	378.8		
		5	540B	0.506	40.1	—	62.1	20.0	$\phi_{Ho}-\Delta p$	107.2	126.5	148.7	217.7	301.5	392.0		
		4	560	0.508	40.1	—	82.1	40.0	$\phi_{Ho}-\Delta p_0-\Delta p-h$	106.2	125.5	147.6	216.2	292.0	380.2		
		5	560B	0.500	39.8	—	81.8	40.2	$\phi_{Ho}-\Delta p$	110.0	129.9	152.6	222.6	301.7	391.7		
		5	575	0.500	75.0	—	81.8	5.0	$\phi_{Ho}-\Delta p_0-\Delta p-h$	110.6	130.1	152.5	224.6	300.5	393.7		
		5	570	0.500	70.0	—	81.8	10.0	$\phi_{Ho}-\Delta p_0-\Delta p-h$	110.4	129.9	152.2	224.6	301.2	392.7		
	4 with flanges; 5 without flanges	3	580	0.499	80.0	160.45	83.0	—	$\Delta T_{met}-\phi_{Ho}-\Delta p_0-\Delta p-h$	110.7	135.7	152.3	228.3	278.8	391.1		



EL. 505
D=0.504 cm
P₀≃71.5 kg/cm² a
L 5 cm
G = 108.2 g/cm sec
G = 128.0 "
G = 149.9 "
G = 218.7 "
L 4.9 cm
(●) G = 218.7 "
G = 297.6 "
G = 386.3 "
Φcr
[W/cm²]
700
600
500
400
300
200
100
0
−0.1
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
Xin

Figure 2



Φ_{cr}
$\left[\frac{W}{cm^2}\right]$
700
600
500
400
300
200
100
0
-0.1
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
X_n
EL. 505
D=0.504 cm
$P_o \simeq 71.5\ kg/cm^2a$
L 5 cm
G = 108.2 g/cm^2sec
G = 128.0 "
G = 149.9 "
G = 218.7 "
L 4.9 cm
G = 218.7 "
G = 297.6 "
G = 386.3 "

Figure 3

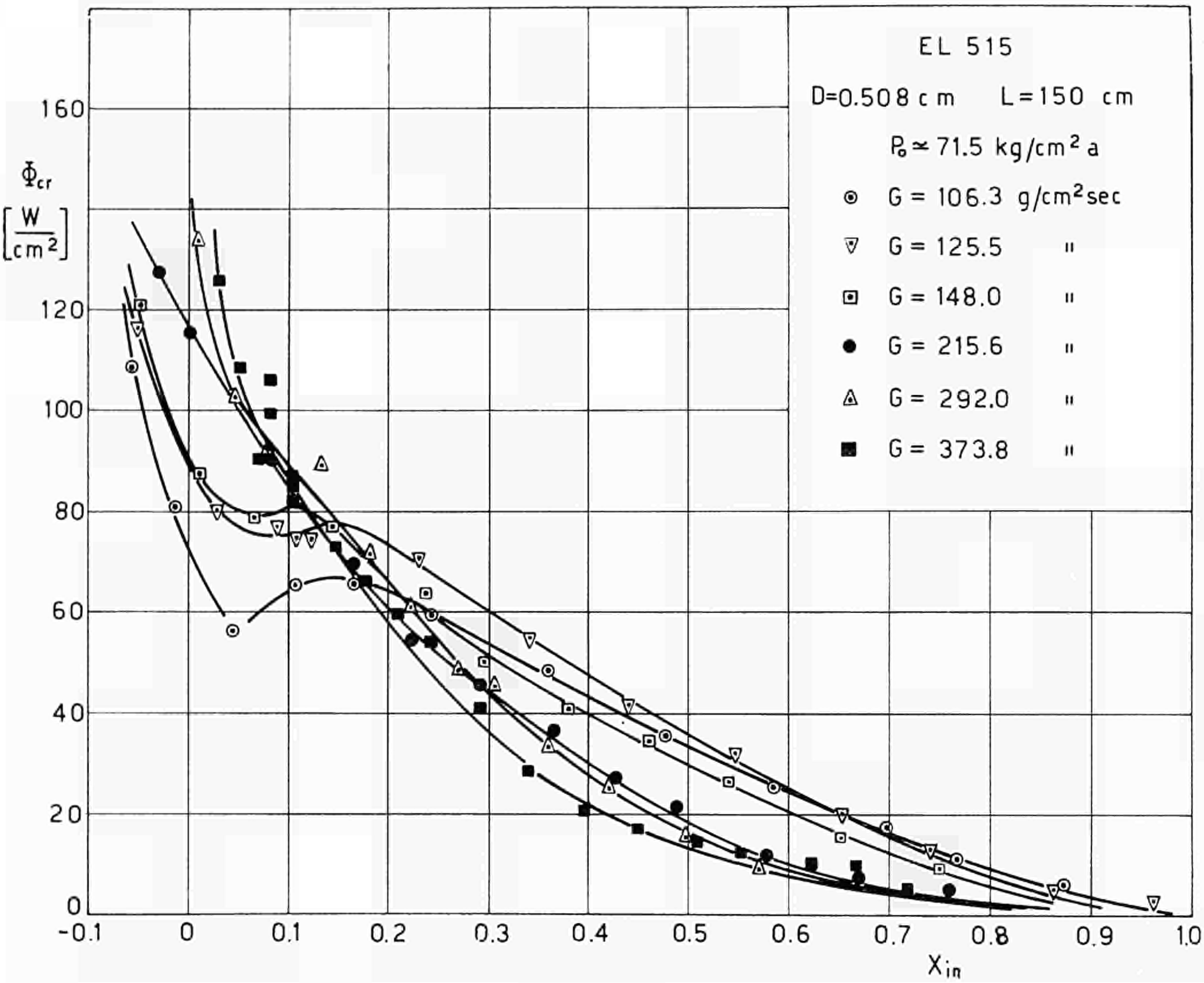


Figure 4a

overall length (80 cm) and different non heated portions (between 0 and 40 cm) clearly shows that, in the case of a succession of heated and non heated portions, the total critical power is rather independent on the non uniformity of the heating (Figure 8).

The preceding ϕ_{cr}, X_{in} plots (Figures 2, 4, 5, 7) present an "instability" on ϕ_{cr} which manifests itself as a resonance-like well, the bottom of which corresponds to inlet qualities equal to 0 or very small. This fact was found to be a general trend of the critical heat flux (or critical power) in function of X_{in} as long as the mass velocity remains inferior to about 200-250 g/cm²s. These "instabilities" seem to depend on the upstream circuit as shown in figure 9: the introduction of a number of orifices at the inlet of the test section eliminates this "instability" (curve C).

A typical plot of frictional pressure drops in adiabatic conditions vs. X is represented in Figure 10.

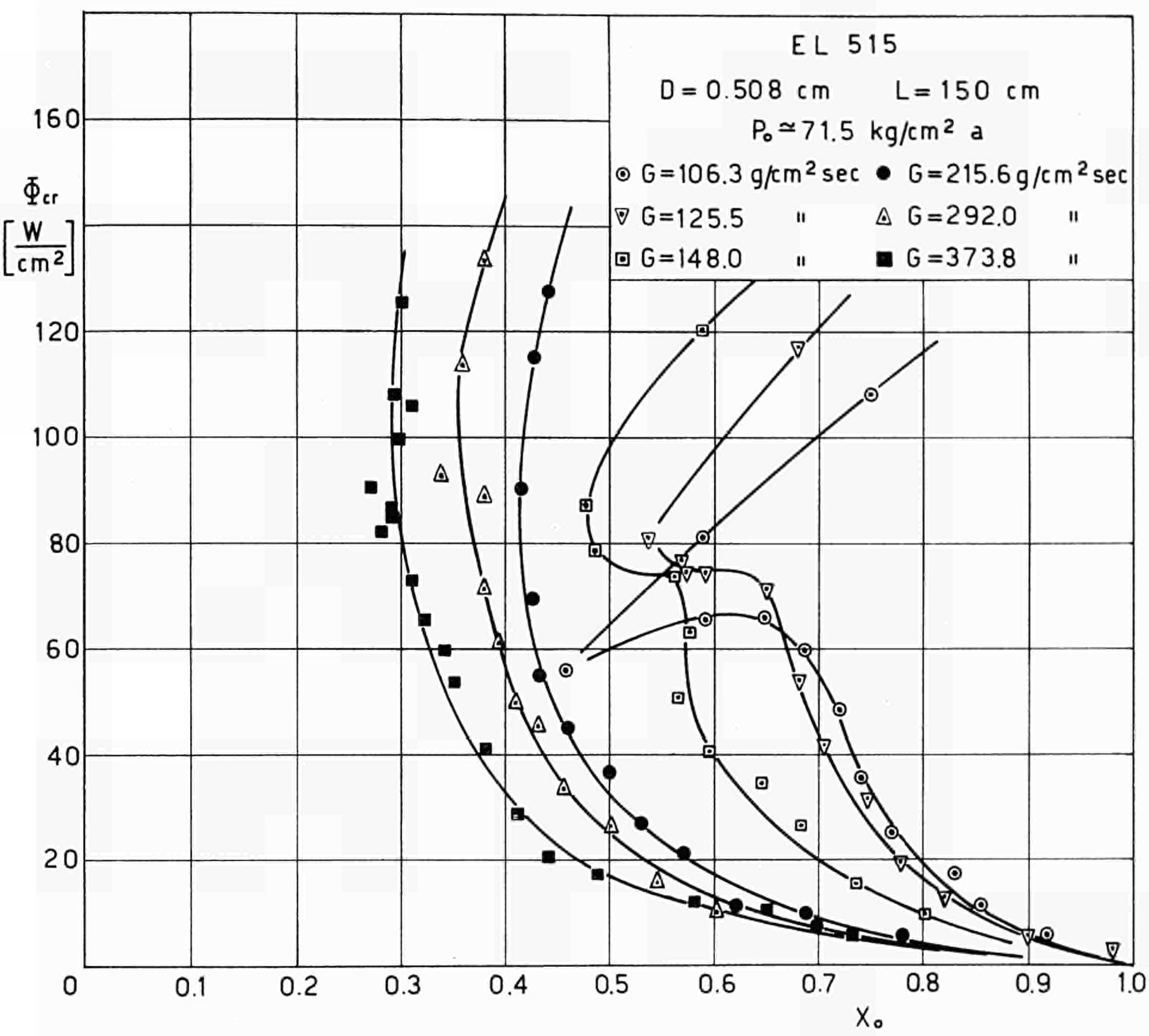


Figure 4b

Decreasing the power from the critical conditions, the wall to bulk temperature difference becomes substantially lower than that obtained during the power increase up to the crisis. This hysteresis in heat transfer coefficient below the crisis is represented for a typical case in Figure 11.

2. HYDRODYNAMICS

The work accomplished at CISE on the hydrodynamics of adiabatic two-phase flow aims at contributing to the general understanding of the so called "annular-dispersed" flow. Most of the experiments were performed with argon-water mixtures at various pressures and at room temperature. Since the physical properties of the liquid phase are different from those of saturated water at temperatures suitable for power reactor

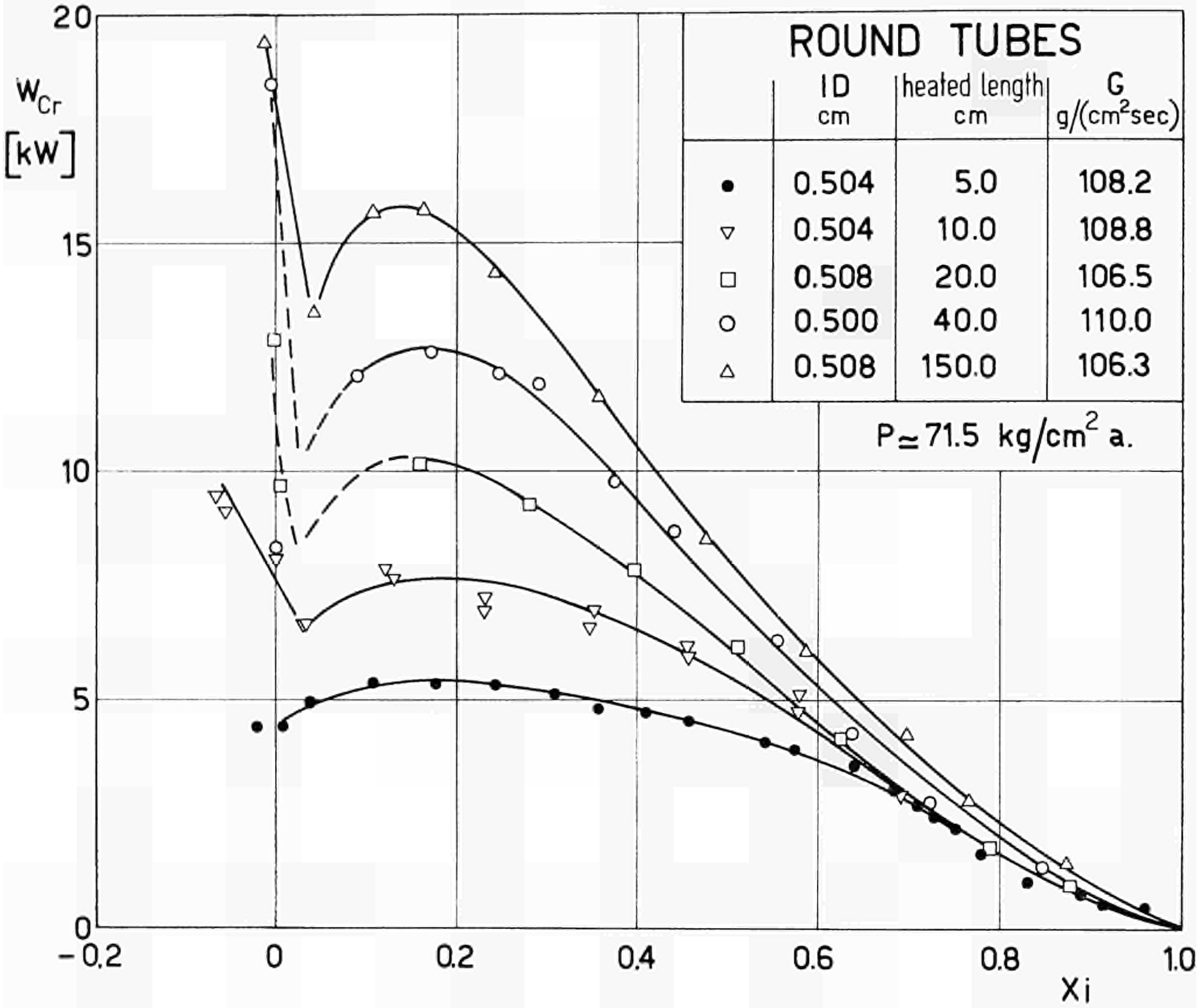


Figure 5

cooling, some experiments were devoted to the investigation of the influence of the physical properties of the liquid phase on the most interesting quantities.

The following quantities were measured:

a) Phase and velocity distribution

b) Pressure drop

c) Mean density of the mixture

d) Pressure oscillations.

a) *Phase and velocity distribution*

In the study of two phase dispersed flow, the region in which the local gas volume fraction is almost zero, that is the liquid film on the wall, is of considerable interest.

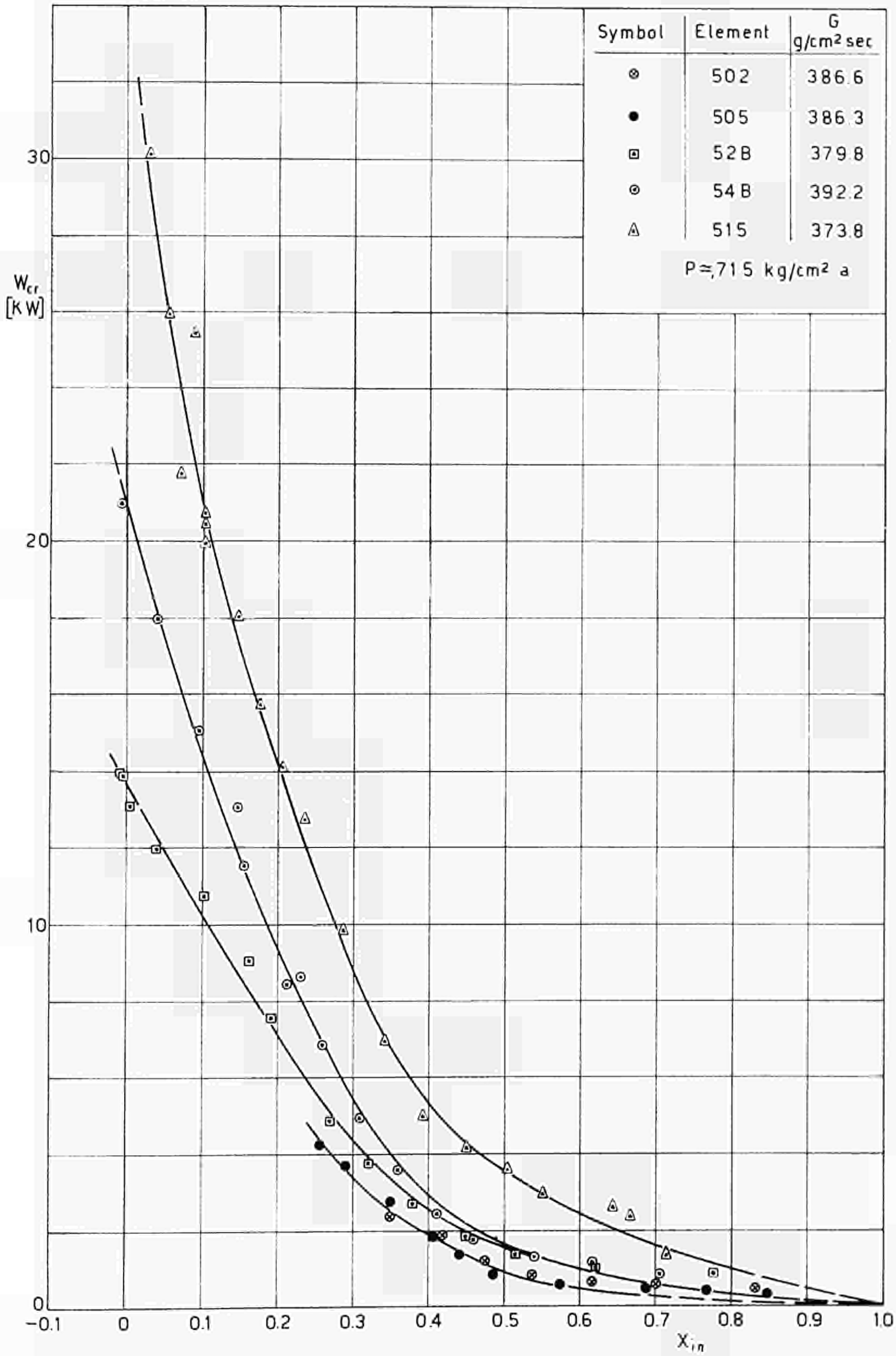

Symbol
Element
G g/cm² sec
502
386.6
505
386.3
52 B
379.8
54 B
392.2
515
373.8
P ≈ 71.5 kg/cm² a
W_cr [KW]
30
20
10
0
−0.1
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
X_in

Figure 6

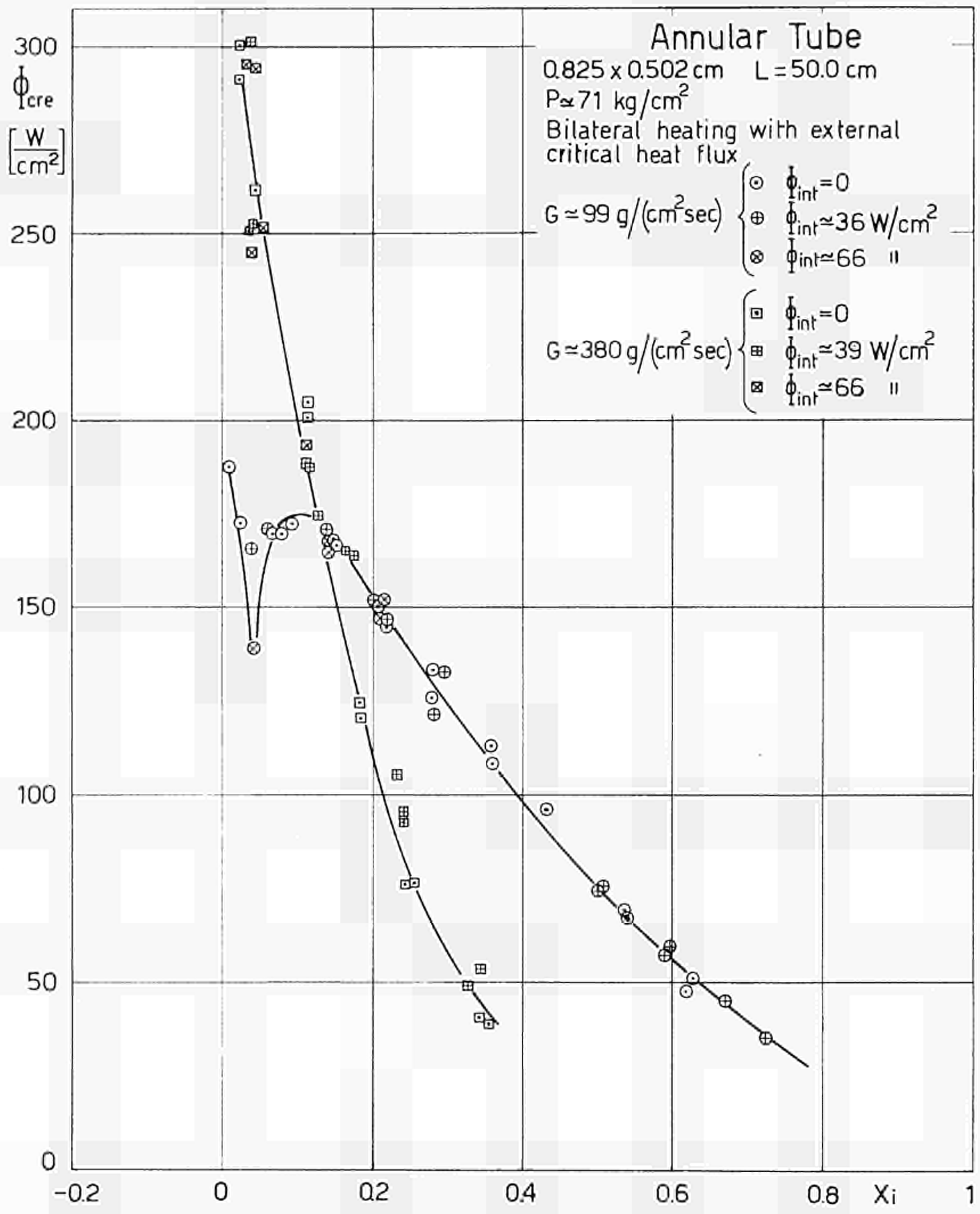
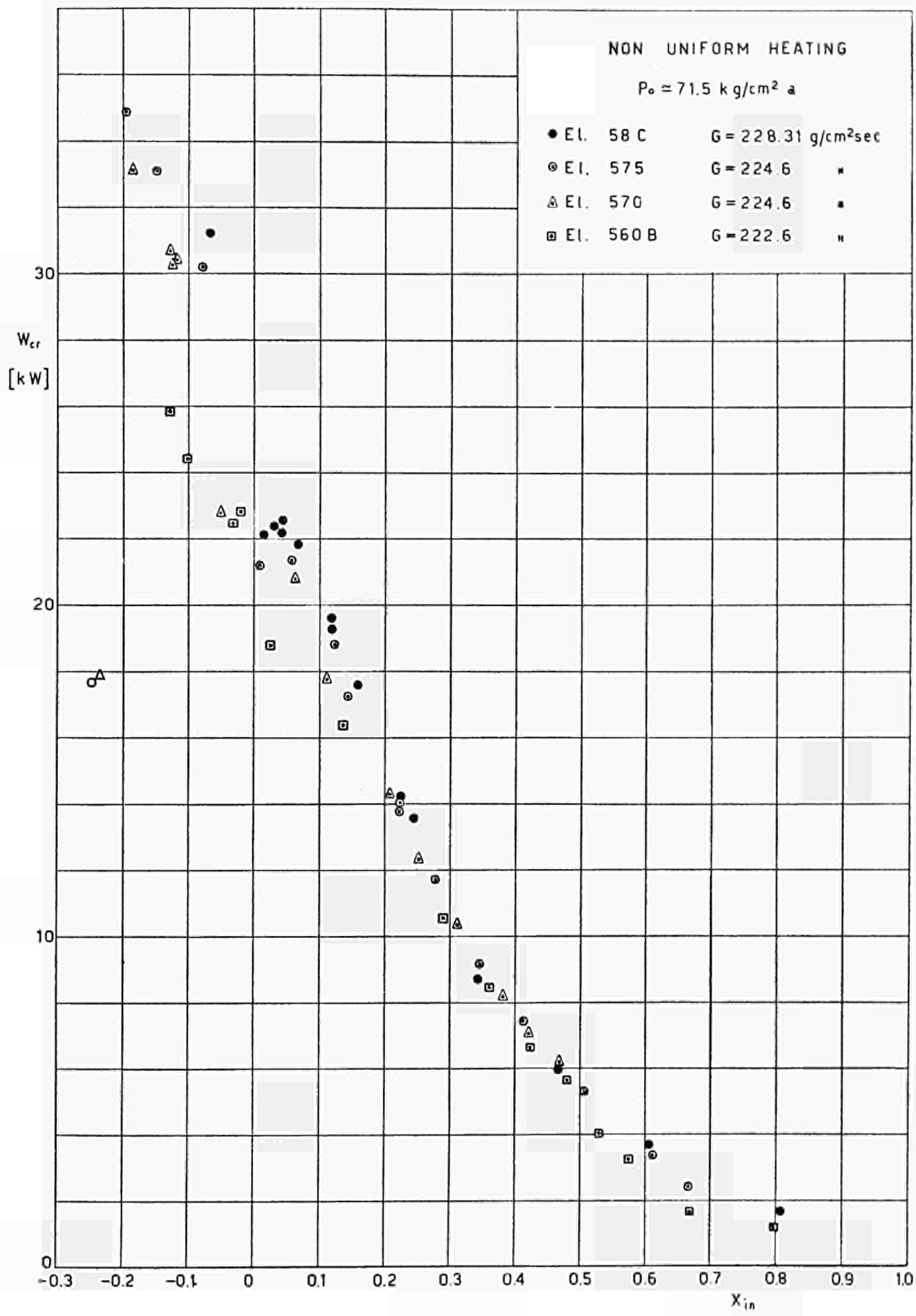

Annular Tube
0.825 x 0.502 cm L = 50.0 cm
P ≃ 71 kg/cm²
Bilateral heating with external critical heat flux
G ≃ 99 g/(cm² sec)
Φint = 0
Φint ≃ 36 W/cm²
Φint ≃ 66 "
G ≃ 380 g/(cm² sec)
Φint = 0
Φint ≃ 39 W/cm²
Φint ≃ 66 "
Φcre
[W/cm²]
300
250
200
150
100
50
0
-0.2
0
0.2
0.4
0.6
0.8
Xi
1

Figure 7



NON UNIFORM HEATING
$P_o \simeq 71.5$ kg/cm² a
● El. 58 C G = 228.31 g/cm²sec
⊙ El. 575 G = 224.6 "
△ El. 570 G = 224.6 "
⊡ El. 560 B G = 222.6 "
W_{cr} [kW]
X_{in}
0
10
20
30
-0.3
-0.2
-0.1
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0

Figure 8

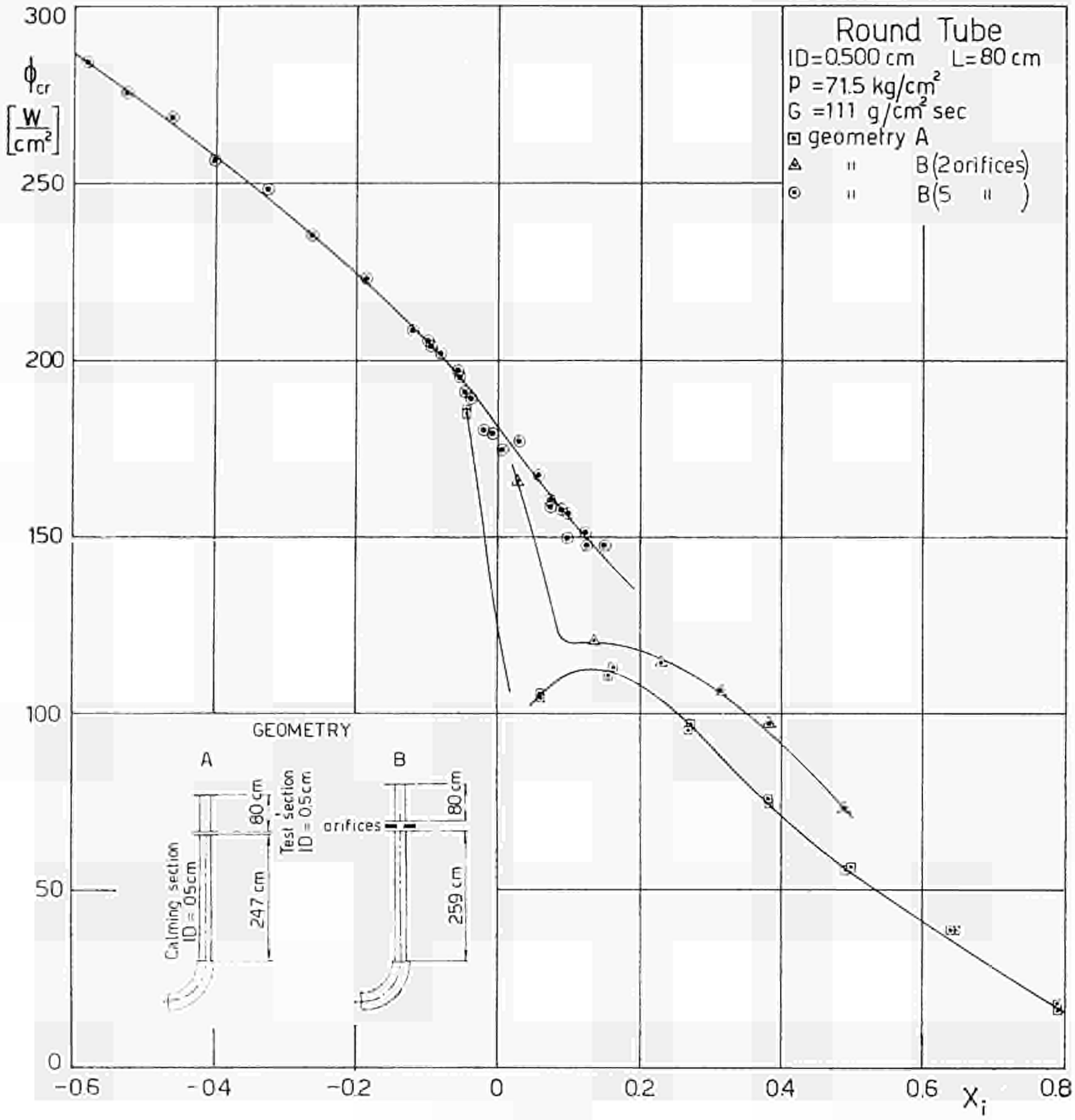

Round Tube
ID=0.500 cm L=80 cm
P =71.5 kg/cm²
G =111 g/cm² sec
geometry A
" B(2 orifices)
" B(5 ")
300
250
200
150
100
50
0
-0.6
-0.4
-0.2
0
0.2
0.4
0.6
0.8
GEOMETRY
A
B
80 cm
Test section ID = 0.5 cm
Calming section ID = 0.5 cm
247 cm
orifices
259 cm

Figure 9

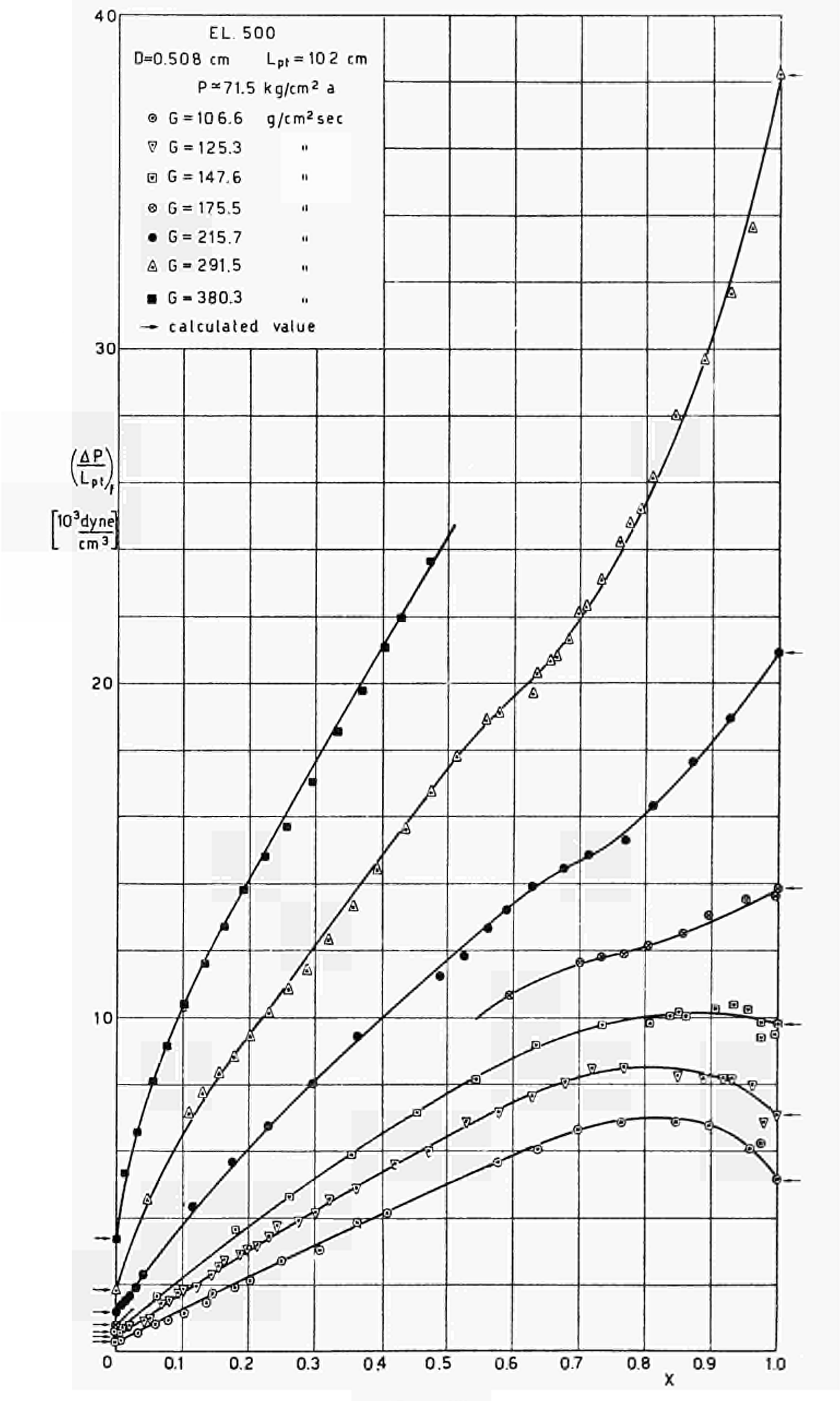
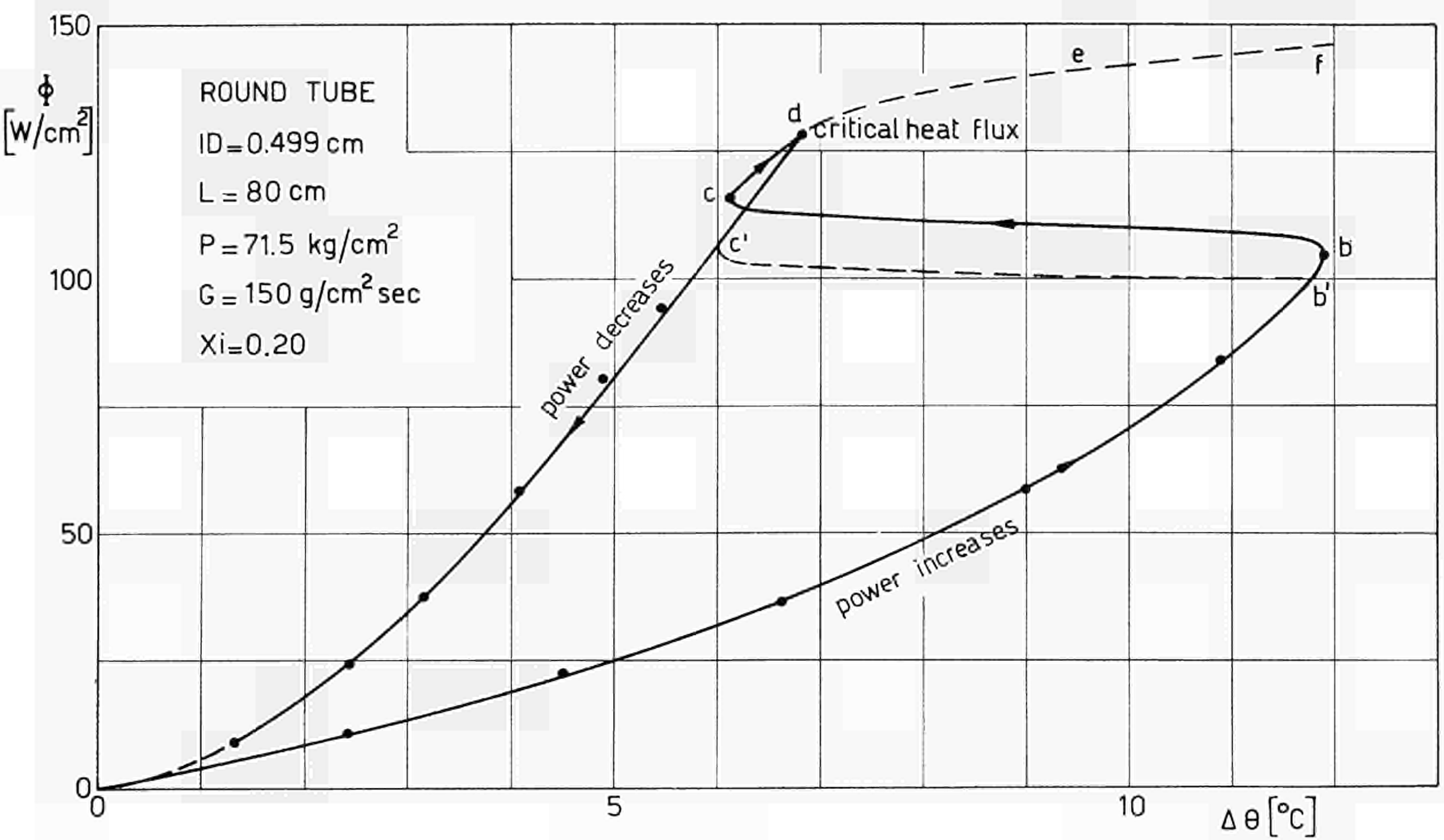

EL. 500
D=0.508 cm   L_pt = 102 cm
P ≃ 71.5 kg/cm² a
G = 106.6 g/cm² sec
G = 125.3 "
G = 147.6 "
G = 175.5 "
G = 215.7 "
G = 291.5 "
G = 380.3 "
calculated value
(ΔP/L_pt)_f
[10³ dyne/cm³]
X
0
10
20
30
40
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0

Figure 10



150
100
50
0
Φ [W/cm2]
ROUND TUBE
ID = 0.499 cm
L = 80 cm
P = 71.5 kg/cm2
G = 150 g/cm2 sec
Xi = 0.20
e
f
d
critical heat flux
c
c'
b
b'
power decreases
power increases
0
5
10
Δ θ [°C]

Figure 11

It is therefore convenient to consider this region separately from the core and to measure directly the film thickness (s). The selected method, based on the measurement of the electrical resistance of the film, gives an electrically averaged value of the thickness which, because of the wavy profile of the film-core interface, does not coincide with the geometrically averaged value. Typical experimental values of the film thickness obtained in round test section (25 mm I.D.) with argon-water mixtures at 18°C and 21.8 kg/cm^2 (abs) are reported in Figure 12 as a function of liquid specific mass flow rate. The film thickness increases with increasing water flow rate but decreases with increasing gas flow rate. This effect is prevalent, so that at constant quality the thickness decreases at increasing total flow rate.

The thickness obtained using ethyl alcohol (90% alcohol by wt.) as the liquid phase (surface tension 24 dyne/cm) are generally lower than those obtained with cold water (73 dyne/cm). This effect increases with increasing total mass flow rate and quality (Figure 13). Density and viscosity of ethyl alcohol being close to those of pure water, the difference should be attributed to the influence of surface tension.

The experimental data on film thickness were examined with the aim of finding out a correlation with the basic variables involved: that is geometry, physical properties and flow rates. Various attempts along this line were not successful, so that attention was turned to the use of other not independent variables, which could intervene in determining the thickness of the liquid film. In this connection a variable of primary importance could be the interfacial shear stress (τ_i). If s_0 is the value of film thickness at the point of the transition between annular and dispersed regime for a given quality (which can be computed in semi-theoretical way), one can write:

$$s = s_0 e^{-K\tau_i} \qquad\qquad s_0 = K_1 D \sqrt{\frac{\rho_g}{\rho_l}} \left[\frac{1-X}{X} \left(1 - \frac{2s_0}{D}\right)^2 \right]^{1-\frac{n}{2}}$$

Figures 14 and 15 show that this correlation takes in quite a good account the dependence of the film thickness (s) on the mass flow rate quality (X) and on the interfacial shear stress (τ_i) with argon-water mixture at room temperature at 21.8 kg/cm^2 pressure.

The phase and velocity distribution in the core of the conduit was investigated only with pure water and argon at $\sim$21.8 kg/cm^2 pressure and 18°C ($\rho_g = 36.1 \times 10^{-3}$ g/cm^3). The following quantities were measured along a single diameter of the test section:

— local values of the specific mass flowrate of both phases (G_g^* and G_l^*) with an extraction probe operating under isokinetic conditions;

— the impact pressure (Δp_p), with the said probe operating as a Pitot tube.

The local values of the linear velocities (U_g and U_l) and of the liquid volume fraction ($1-\alpha$) were derived introducing a relationship between the impact pressure, as measured by the probe, and the basic variables involved (see CISE Report R 35):

$$\Delta p_p = \frac{1}{2} \alpha^2 \rho_g U_g^2 + \frac{1+\alpha}{2} (1-\alpha) \rho_l U_l^2$$

Figure 16 gives an example of the quantities measured; the corresponding value of the film thickness is also indicated.

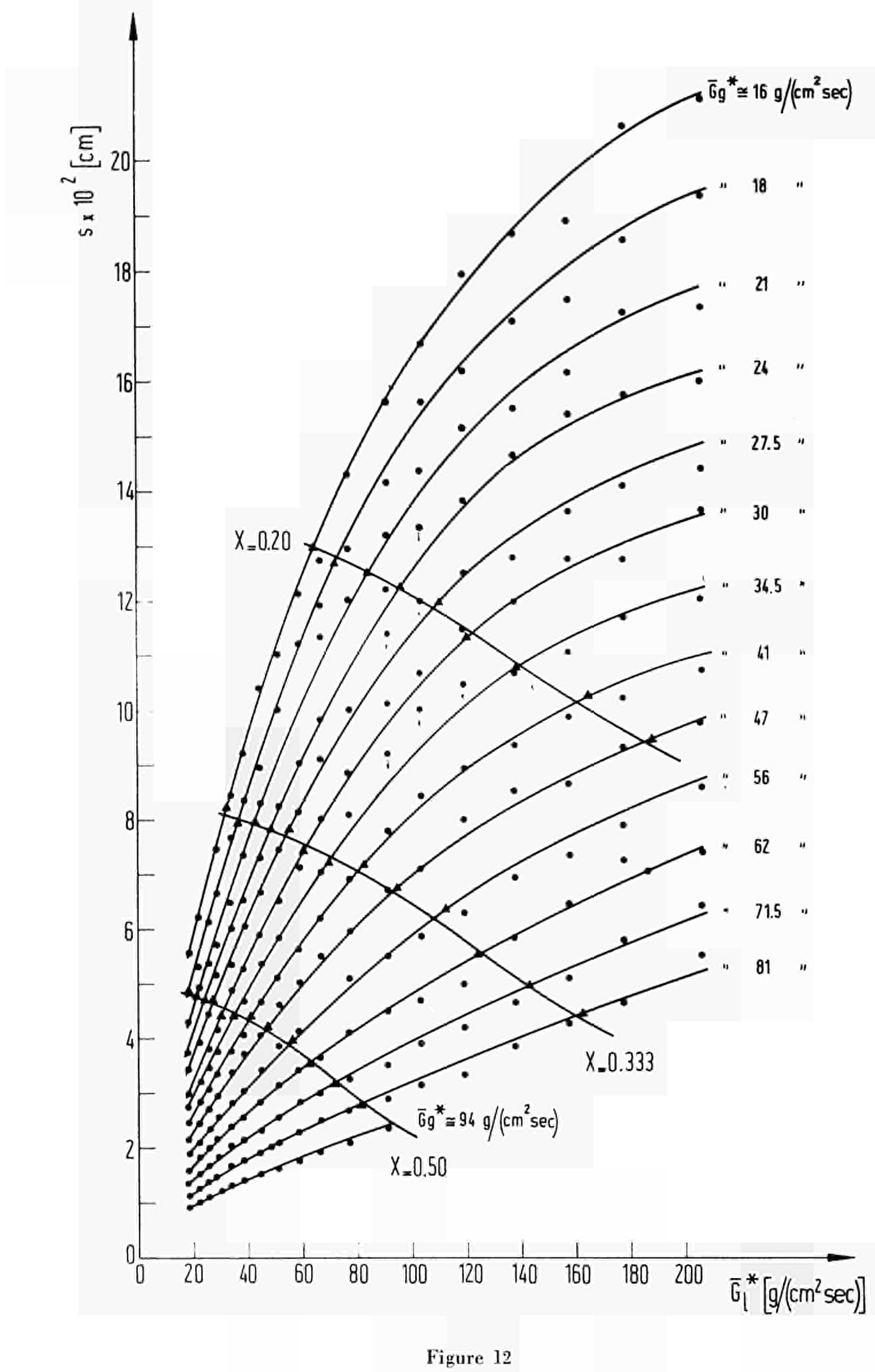


Figure 12

— *Argon-water mixtures at 21.8 kg/cm² and 18°C.*
— *Film thickness vs. liquid specific mass flow rate.*
— *Gas specific mass flow rate and quality as parameters.*

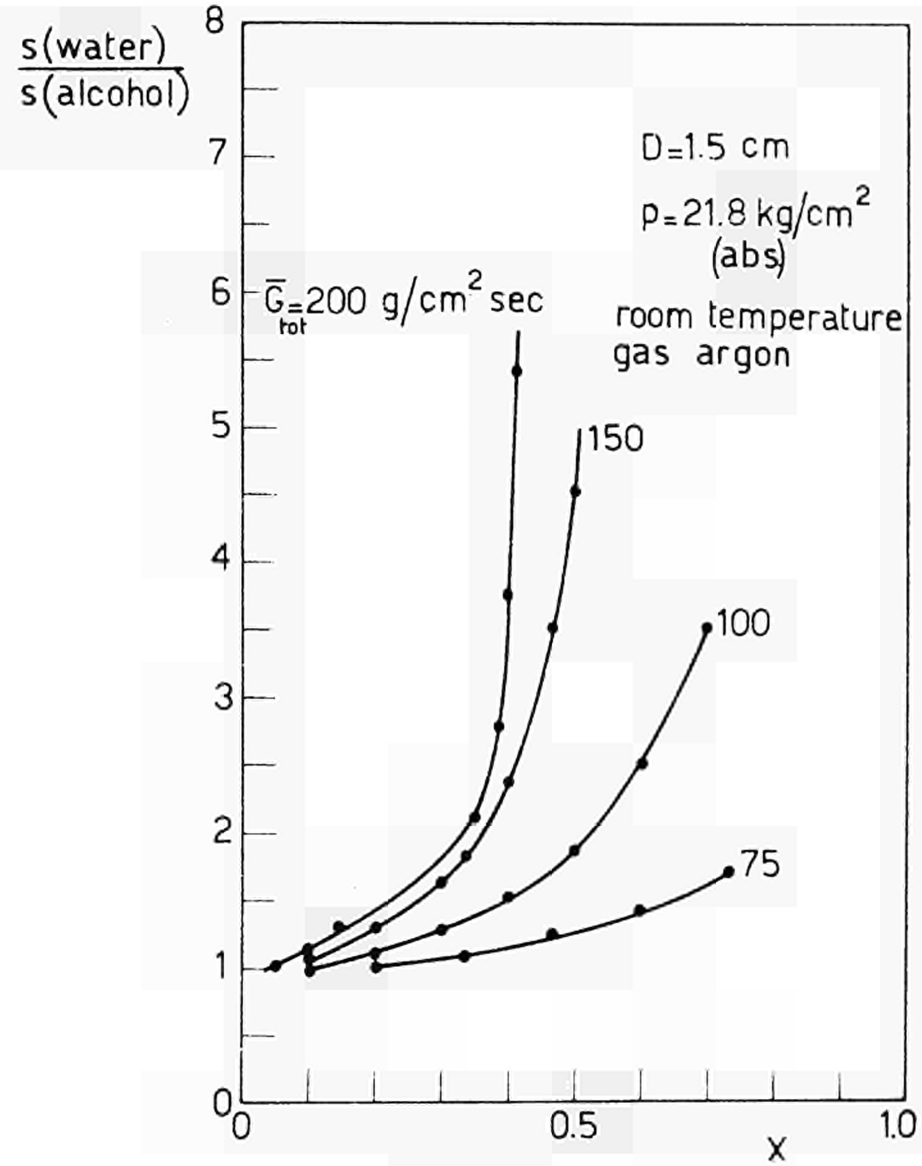


Figure 13

Influence of surface tension on liquid film thickness: ratio of liquid film thickness with water and ethyl alcohol as the liquid phase vs. mass quality at constant total specific mass flow rate.

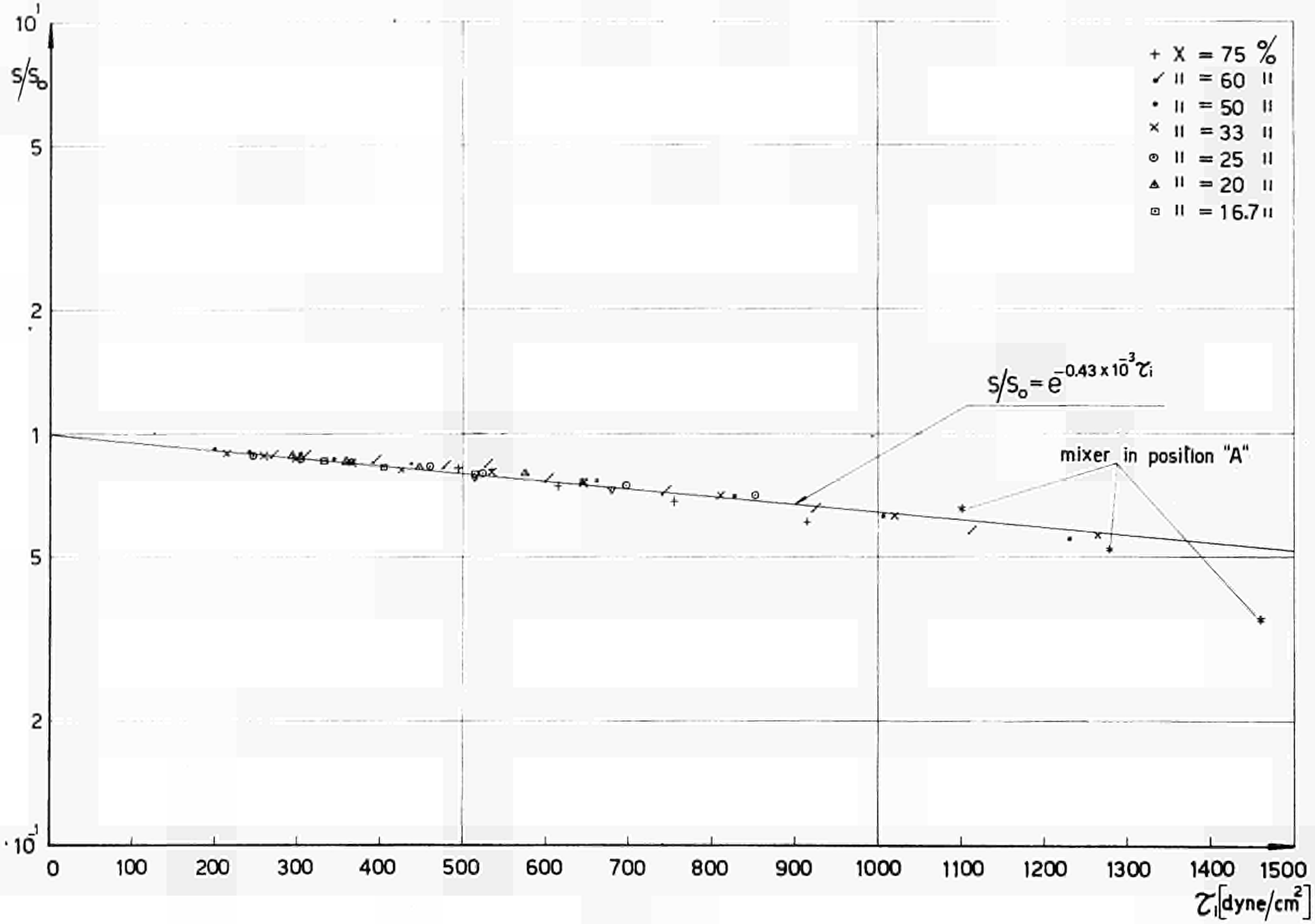


Figure 14

Correlation between the ratio s/s_o and interfacial shear stress at constant mass quality.

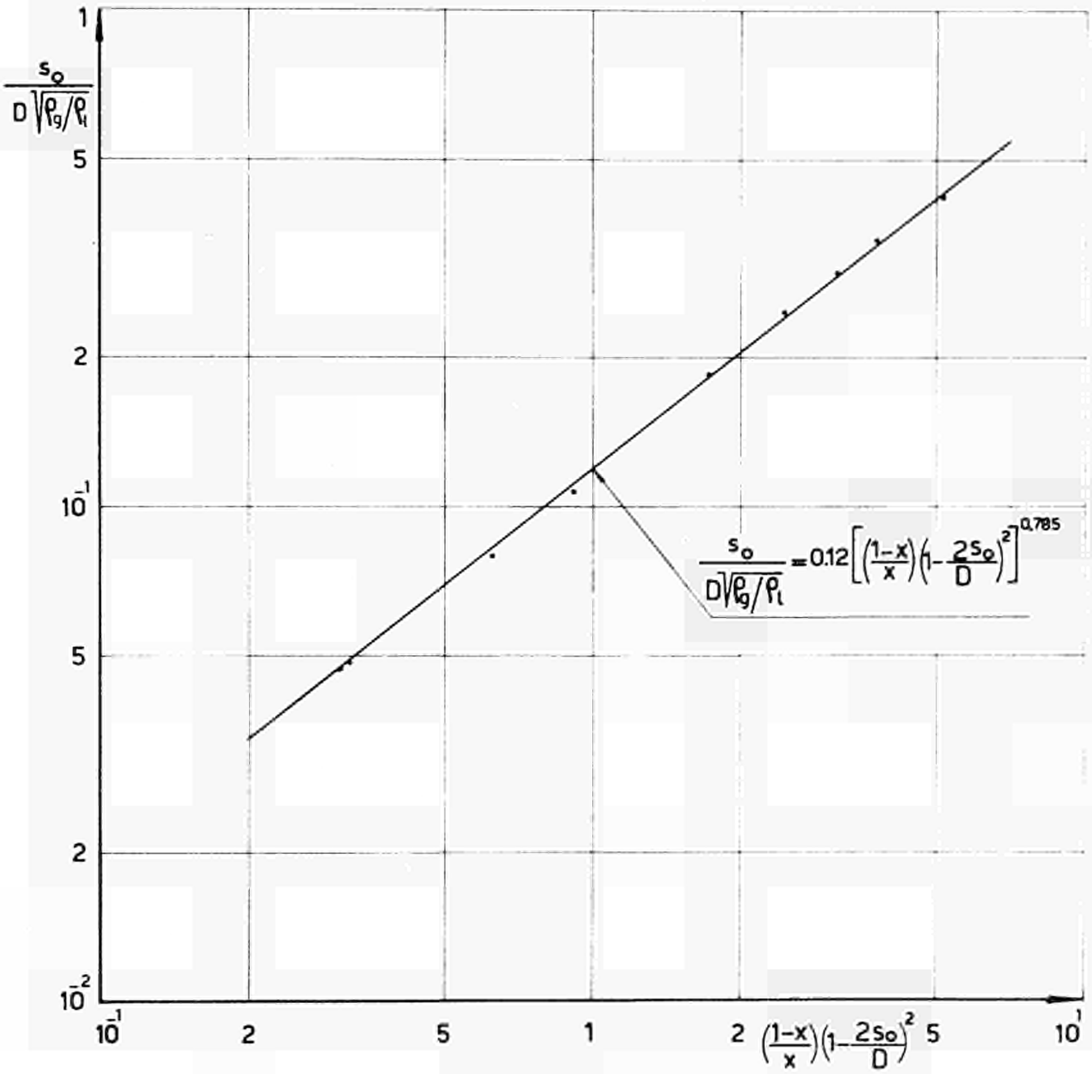


Figure 15

Correlation between $\frac{s_0}{D\sqrt{\rho_g/\rho_l}}$ *and* $\left(\frac{1-X}{X}\right)\left(1-\frac{2s_0}{D}\right)^2$

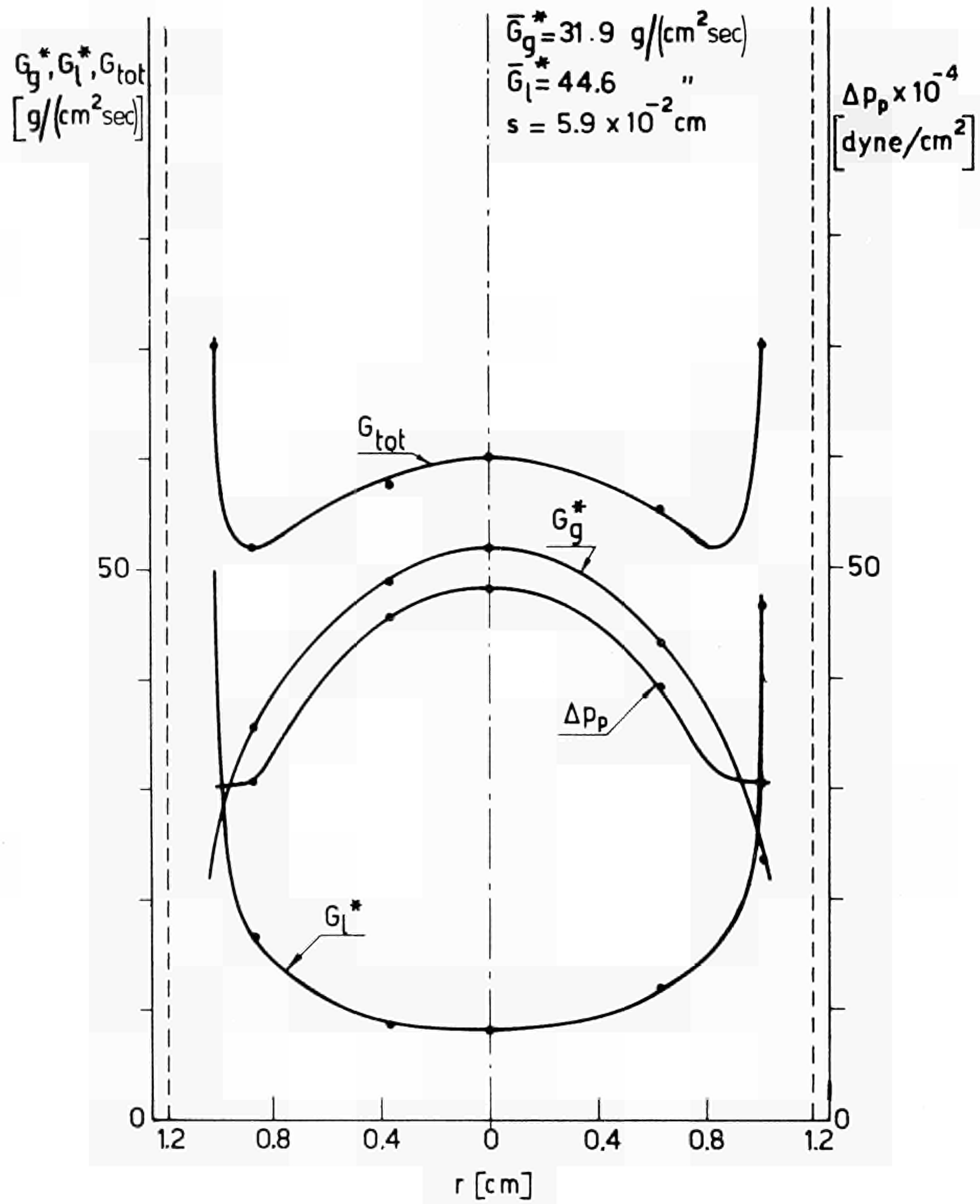


Figure 16

A typical distribution of quantities (G_l^*, G_g^*, G_{tot}, Δp) as measured with the extraction probe.

The computed values of U_g, S and $1-\alpha$ for the same operating conditions are presented in Figure 17. The local slip ratio (S) in the core is close to one and nearly constant along a diameter, with values somewhat higher near the wall. The average linear velocity of the liquid in the core is however remarkably lower than that of the gas, due to the higher concentration of the liquid in the low velocity region. In fact, as a general trend, the local liquid volume fraction $(1-\alpha)$ varies strongly over the conduit cross-section, having a minimum in correspondence with the axis.

b) *Pressure drop*

As stated in our previosu report (CISE R 43), in vertical two-phase flow the pressure loss cannot be measured directly but can be computed, in principle, according to the energy equation of two-phase flow through the difference between the measured total pressure drop and the acceleration plus the head term.

In practice, the evaluation of the acceleration term may present some difficulty. Since in our experiments the variation of the kinetic energy along the flow direction was very small, the acceleration term was neglected and the pressure loss was computed by simply subtracting from the total pressure drop the head term.

This term, in two-phase flow, is not related to the actual mean density of the mixtures $(\bar{\rho} = \bar{\alpha}\rho_g + (1-\bar{\alpha})\rho_l)$. In fact, the power recovery due to elevation of the two fluids is related to the volume flowrates, hence to the flowrate density $(\rho^* = X_v\,\rho_g + (1-x_v)\,\rho_l)$. Following this line, the pressure losses were computed according to the relationship:

$$\left(\frac{\Delta p}{\Delta z}\right)_f = \frac{\Delta p}{\Delta z} - g\,\rho^*$$

The pressure losses computed in the case of argon water mixtures at 18°C and 21.8 kg/cm^2 pressure ($\rho_g = 36.1 \times 10^{-3}$ g/cm^2) are presented in Figure 18. The values corresponding to single-phase flow ($X = 0$ and $X = 1$) are also reported. The following remarks can be made:

— In the high quality region ($X \geqslant 0.95$), pressure losses decrease abruptly towards the values corresponding to single-phase (gas) flow ($X = 1$) at the same total flowrate. In fact at constant total flowrate, above a certain quality, pressure losses are larger than in single-phase flow (gas).

— In the low quality region the trend of pressure loss curve versus mass quality changes at the transition: dispersed flow-bubble or slug flow (dotted lines).

The influence of the surface tension on pressure losses is shown in Figure 19 where the ratio between the pressure loss with water and with ethyl alcohol as liquid phase are plotted vs. mass quality: pressure losses decrease with decreasing the surface tension, this effect being more pronounced at the highest flowrate and mass quality values. The influence of surface tension on pressure losses seems however less pronounced than on film thickness. Anyway it must be kept in mind that frothing might have an effect at the highest values of flowrates and quality, since the liquid used (90% by wt. ethyl alcohol and water) was not pure.

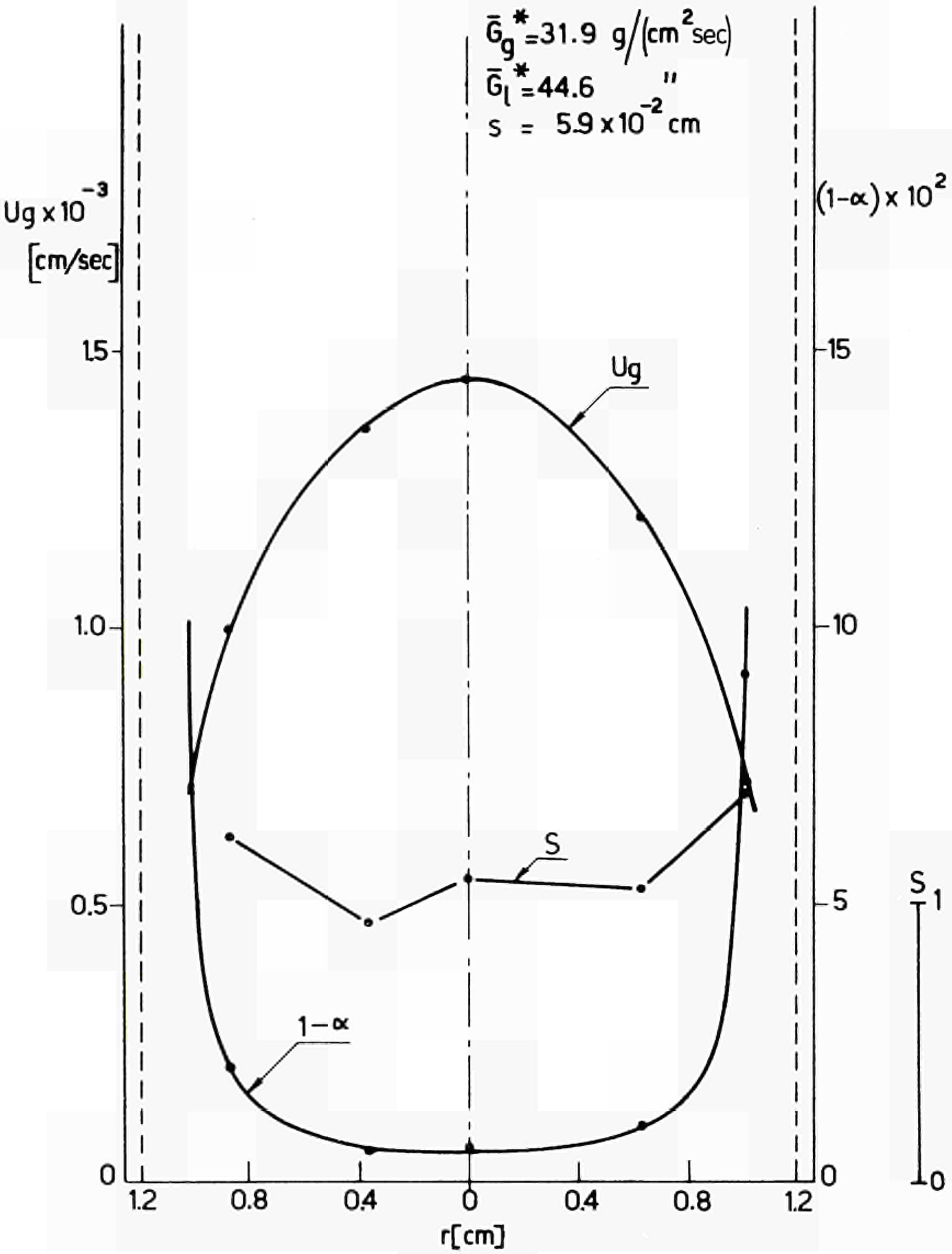


Figure 17

A typical distribution of quantities (U_g, $1-\alpha$, S) calculated from the extraction probe data.

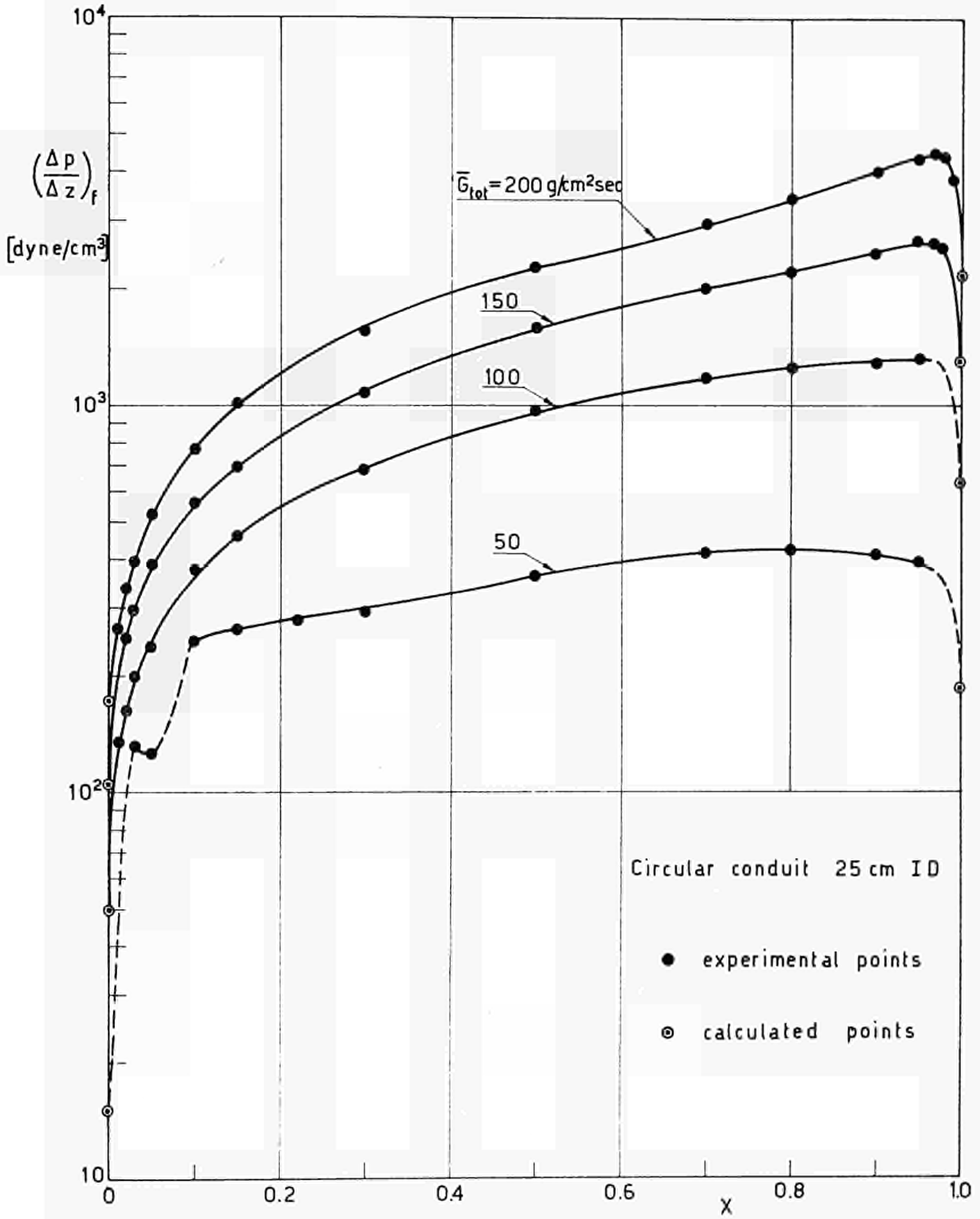


Figure 18

Argon-water mixture at 21.8 kg/cm² and 18°C.
Pressure loss vs. mass quality with total specific mass flow rate as a parameter.

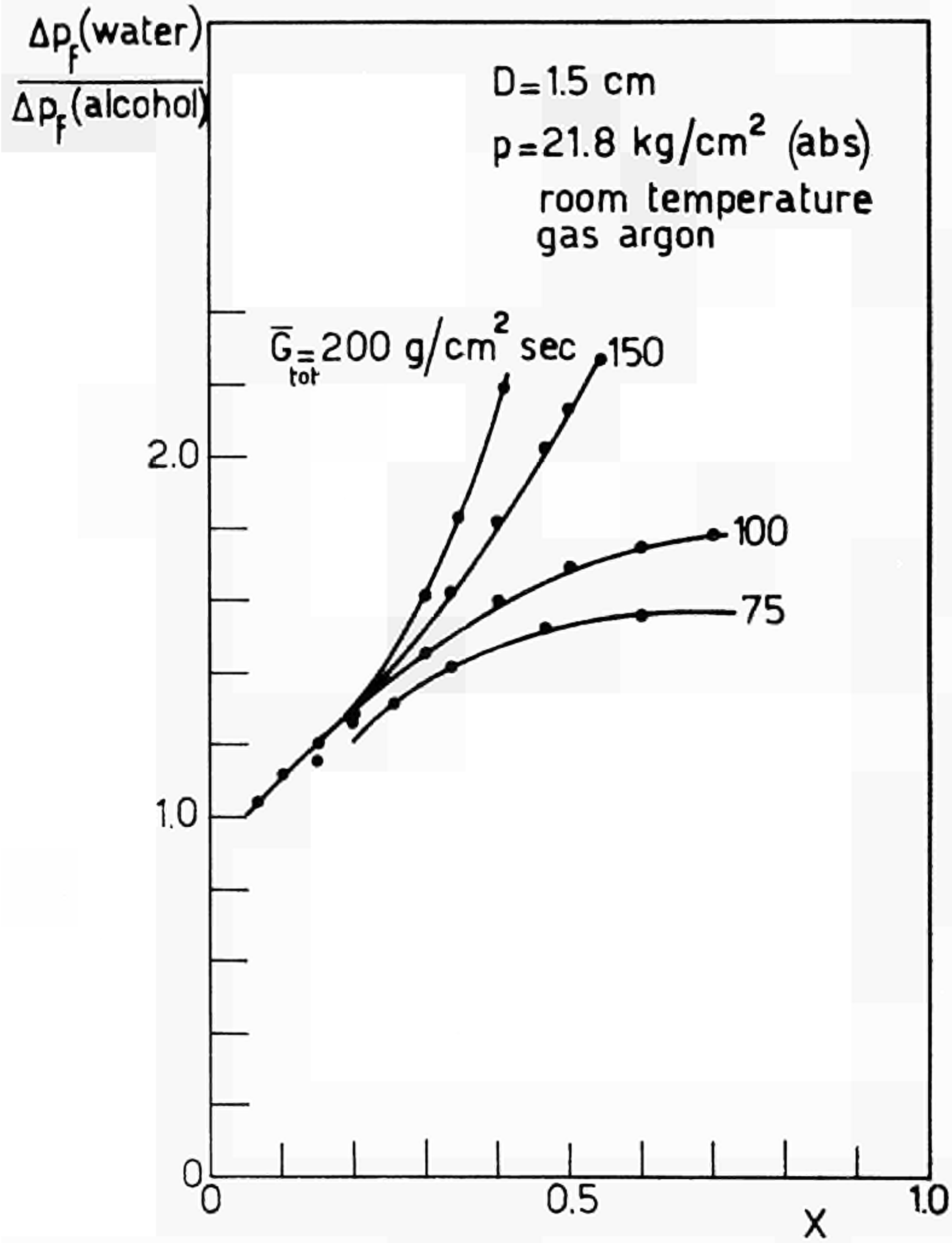


Figure 19

Influence of surface tension on pressure loss: ratio of pressure loss with water and ethyl alcohol as the liquid phase vs. mass quality at constant total specific mass flow rate.

c) *Mean density of the mixture*

The knowledge of the mean density of two-phase mixtures is of primary importance for its implications in the nuclear design of reactors cooled by such mixtures. The determination of the mean density was carried out in two ways:

— from the overall liquid volume fraction $(1-\bar{\alpha})$ values obtained through an integration process from the local values of $(1-\alpha)$ in the core and the value of film thickness;

— by a direct measurement through a beta-ray attenuation method. For this purpose a small source was constructed, which could be placed in different positions along the duct diameter thus allowing density measurements in a non-uniform absorbing medium such as in the present case. The experimental results are in a reasonable agreement with those obtained as said before (Figure 20).

d) *Pressure oscillations*

The oscillations of the static pressure in the test section were investigated by means of a low inertia pressure transducer (strain gauge type) electrically connected to an oscilloscope, the traces of which were recorded photographically.

The amplitude of the oscillations is always quite moderate with respect to the line pressure, even if considerably higher than in single-phase (gas) flow. The main frequency of the oscillations in "annular-dispersed flow" (Figure 21) is comprised between $10 \div 50$ c/sec. No significant oscillations were detected, even during the transition between the different regimes. The main frequency however of the oscillations in the plug regime was remarkably lower (1 c/sec) than that corresponding to the dispersed regime (Figure 22).

— Argon-water mixture at 21.8 kg/cm² and 18°C circular test section 2.5 cm I.D.

— Comparison of mean density values obtained with the extraction probe and the beta probe measurements.

$\overline{G}_g$* (g/cm²sec)	$\overline{G}_l$* (g/cm²sec)	$\bar{\rho}$ beta (g/cm³)	$\dfrac{\bar{\rho}\text{ beta}}{\rho\text{ probe}}$
24.4	25.7	0.141	0.78
24.4	44.6	0.187	0.82
24.4	77	0.256	0.80
24.4	134	0.292	0.73
31.9	25.7	0.130	0.84
31.9	44.6	0.162	0.84
31.9	77	0.222	0.86
54.1	25.7	0.097	1.00
54.1	44.6	0.118	0.94
54.2	77	0.133	0.87
54.4	134	0.203	0.77
72	25.7	0.078	1.04
72	44.6	0.090	0.90
72	77	0.109	0.94
72.5	134	0.180	0.93

Figure 20

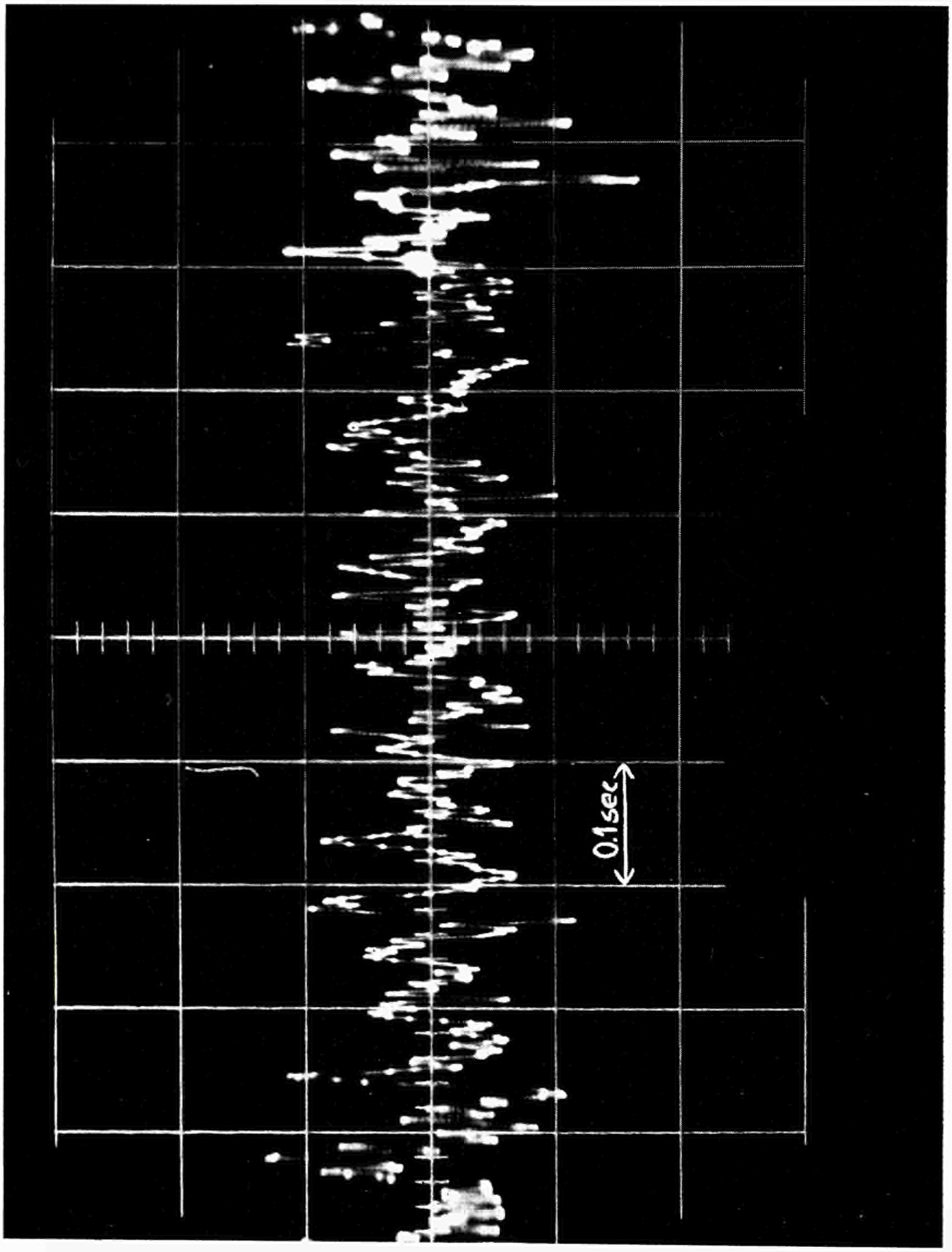


Figure 21

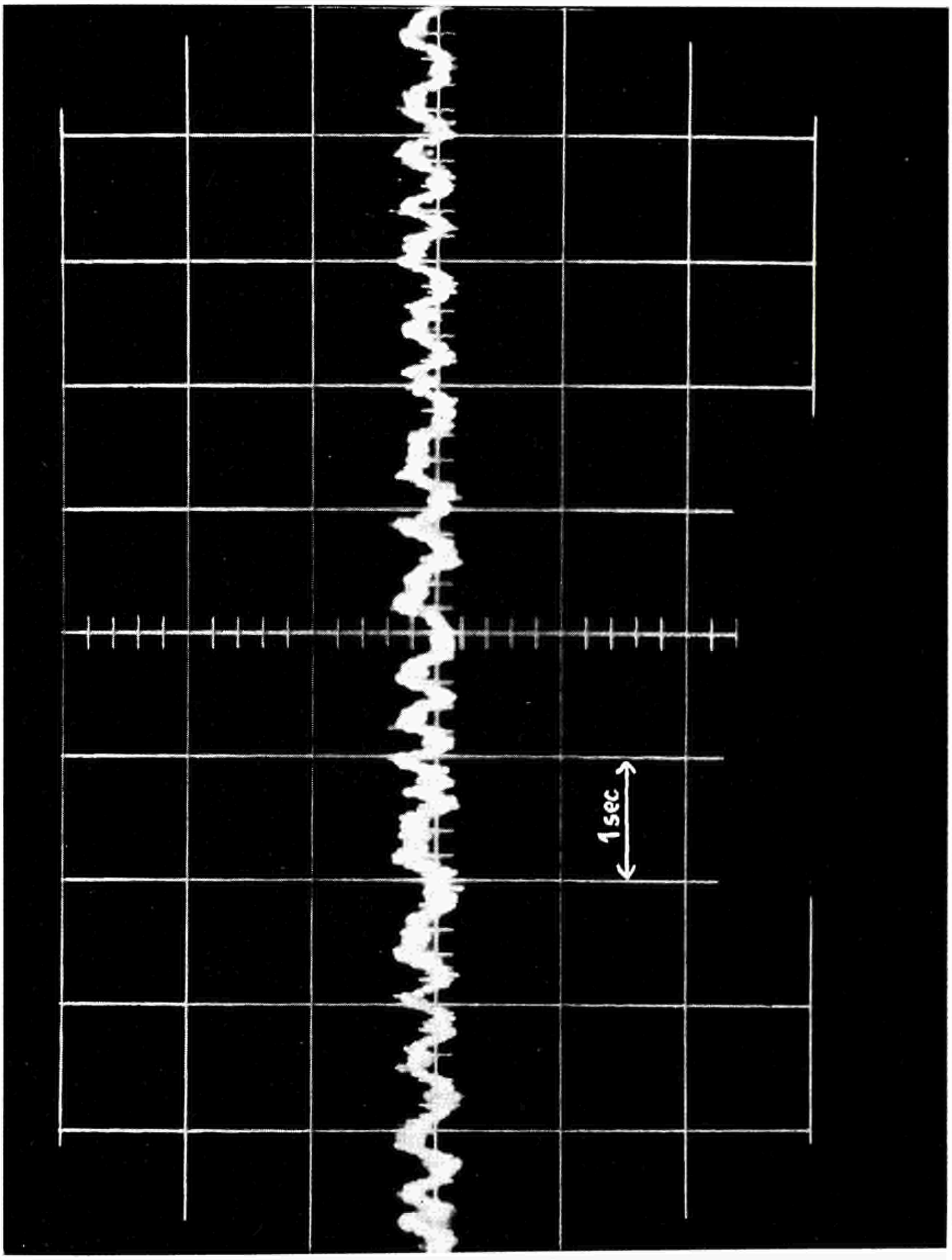


Figure 22

III. GENERAL DISCUSSION ON BURNOUT HEAT FLUX

Chairman: Pr. W. M. ROHSENOW

The group boldly began discussions of the criteria for and definition of burnout heat flux. This led to discussions of stability and flow regimes.

BURNOUT HEAT FLUX

After a few moments of lip service to deploring the use of such terms as burnout, DNB, etc., the group went to a discussion of the phenomena which I shall here call critical flux, realizing that we really mean some point where the wall temperature rises more rapidly than previously as some quantity such as flow rate or q/A or pressure is changed.

It was pointed out that in some apparatus the critical condition is obtained by decreasing flow rate and in others by increasing q/A. It was also pointed out that most critical conditions are detected by electric resistance change over a large area. Some detection apparatus respond to local critical conditions. These probably occur at lower heat flux conditions than do the area detection apparatus.

There seemed to be general agreement that there were really two regions of interest. In one, the subcooled and low quality range, the critical heat fluxes were large and generally resulted in wall temperature rises sufficient to cause destruction of the wall surfaces. In the other, the higher quality range, there resulted at some heat flux a more rapid rise in wall temperature which may or may not be dangerous.

In this latter region it was tentatively suggested that a particular ΔT rise in wall temperature or absolute magnitude of wall temperature be established as the criterion for $(q/A)_{crit}$. This suggestion was voted down because the dangerous wall temperature might be quite different for various reactor designs. There was widespread agreement that in the higher quality region the data be presented as q/A vs. ΔT or h vs. ΔT for no oscillations in flow or pressure. However, where possible, the amplitude in the wall temperature oscillations should be given on the h vs. ΔT plot.

A discussion of stability brought out various kinds of instability that occurs in our test systems:

a) Natural circulation loop instability.

b) Parallel channel natural circulation instability.

c) Parallel channel forced circulation instability.

d) Static vs. dynamic instability.

It was made clear that the loop or system instability causes in most cases, premature critical conditions. The only type of critical (q/A) we can really hope to correlate is the kind that occurs in the absence of flow and pressure oscillations—only wall temperature oscillations being present. Critical conditions occuring with flow oscillations are associated primarily with a particular system geometry and influenced by practically every component of the system.

CONCLUSION

Research on two-phase flow has undergone a considerable expansion over the last few years. This development indubitably reflects the influence of the problems raised by boiling- or pressurized-water reactor technology, while the significance of these questions is accentuated by the attention at present focussed on reactors cooled by steam-water mixtures and by liquid metals. Furthermore, the boiling phenomenon is itself an extremely complex matter which, as this conference has demonstrated, raises a wide variety of problems.

I do not think it is necessary to insist on the usefulness of such a meeting, which provides the experts with an opportunity to compare opinions and to exchange notes on the results which they have obtained. This comment applies even more since the overall programs are coordinated between AEC and EURATOM.

Generally speaking, as far as the research in progress is concerned, it is not expected to arrive at an immediate understanding of all the aspects of the problem. What we want to do is to arrive at provisional solutions and then to go on subsequently to more sophisticated ones. For example: as slug flow is a source of flow instability, we might consider by-passing it by methods such as vortex flow or steam-water mixtures. A complete understanding of the phenomenon will come later.

At the same time, other studies are more specifically directed at gaining a grasp of the laws governing the various problems. These two methods are both necessary and are complementary to each other. This is one of the conclusions of the meeting.

It would be very desirable for such discussions to be arranged at regular intervals in the future. It may be an idea to pay less attention to the programs themselves, since they will be known, and to give more time to presentation of the results. Thus it may be necessary to re-arrange the papers so that they are given in a different sequence according to subject.

In conclusion of the meeting, a short debate was held on the measurement methods and on a definition of some of the more controversial terms. The suggestion was also put forward that a kind of catalogue of the measuring instruments which might be used for two-phase flow be drawn up: on the one hand, hydrodynamic measurements and on the other hand wall temperature measurements. The catalogue would include the characteristics of the devices, drawings, conditions of use and areas of use. Proposals on this point can be put forward now.

The discussion of the exact definitions of terms at present employed to describe certain phenomena could be continued in the next meeting.

R. MORIN