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TWO-PHASE FLOW STUDIES IN JAPAN

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Abstract—This article reveiws recent Japanese work on gas–liquid two-phase flows in 14 classified fields of research, covering 82 organizations or laboratories in Japan, both at universities and in industry, where two-phase flow studies are being pursued. At the end of this paper, the organizations or laboratories, correspondents and research subjects are listed for information together with illustrative papers, up to 4 per organization or laboratory.

Key Words: review, current research projects, gas–liquid flow

1. INTRODUCTION

The present review was originally prepared for presentation at the *Japan–U.S. Seminar on Two-phase Flow Dyanmics*, held in Ohtsu, Japan, during 15–20 July 1988, co-sponsored by the Japan Society for the Promotion of Science and the U.S. National Science Foundation. All of the data included in this review were collected on the basis of a request for information sent to more than 100 organizations or laboratories in Japan doing work in the field of gas–liquid two-phase flow. Solid–liquid and solid–gas two-phase flow studies are excluded from this review. The response to the request was very positive and a huge amount of information was supplied by 82 organizations or laboratories in written form, together with published papers relating to their current research projects.

Table 1 shows separately the number of organizations or laboratories involved in each of the 13 classified fields of research in the area of gas–liquid two-phase flow. All these researches or development works are being performed either at universities or in industry. This table reflects the fact that the physical phenomena taking place in the production of various materials and manufacturing processes, as well as in various systems of energy conversion and energy utilizaiton, should be better understood to improve the reliability of products, to elevate productivity, to lower energy consumption, to enhance the thermal efficiency and safety of the systems etc. It is pointed out here that table 1 lists only part of the gas–liquid two-phase flow activities currently underway in Japan, and there are still many related works which are not covered by this review. In addition, there are several academic societies working very actively on gas–liquid two-phase flow, e.g. the Japan Society of Mechanical Engineers, the Heat Transfer Society of Japan, the Atomic Energy Society of Japan and the newly-established Japan Society of Multiphase Flow. The Committee on Numerics of Nuclear Reactor Thermal-hydraulics, which is one of the special committees belonged to the Atomic Energy Society of Japan, comprises about 80 active members from universities, governmental research institutes, industry and computation companies. This committee has published very good surveys on two-phase thermal-hydraulic problems which should be solved in relation to nuclear reactor safety (Nariai 1986–1988). These surveys also include state-of-the-art reports on the progress in two-phase flow modeling and code developments in Japan. The Japan Society of Multiphase Flow organizes a multiphase flow symposium every year and presently has two subcommittees dealing with numerical and visual simulations of two-phase flow phenomena.

In what follows, a few examples of current work in Japan in each field shown in table 1 are summarized with some illustrations based on the submission; these have been selected somewhat randomly to outline the various fields instead of showing summaries of all research activities. In the appendix up to 4 illustrative references are listed for each submission.

Table 1. Number of laboratories in each research field

Research field	Number of laboratories
Fundamental equations for two-phase flow and closure relations	7
Flow regime identification, modeling and dynamics	19
Phase separation phenomena	6
Phase distribution phenomena and two-phase flow turbulence	25
Interfacial phenomena	20
Non-equilibrium modeling	5
Vapor explosion	3
Thermal-hydraulic instabilities	19
Pressure and void wave propagation and shock phenomena	10
Critical two-phase flow	7
Forced convective heat transfer (heat transfer enhancement and CHF)	24
Post dryout heat transfer	13
Code development and verification	8

2. FUNDAMENTAL EQUATIONS AND CLOSURE RELATIONS

Because of the rapidly growing need for accurate and reliable numerical analysis of two-phase flow problems, fundamental equations and closure relations for two-phase flow systems have been pursued. They can be classified into two groups:

(1) general consideration of the basic equations; and
(2) basic equations for special applications.

Regarding the first group, work is proceeding at Kyoto University (Kataoka & Serizawa), Himeji Institute of Technology (Nakanishi and coworkers) and the Japan Institute of Nuclear Safety (JINS) (Okazaki) for gas–liquid systems, and at Kobe University (Sakaguchi and coworkers) for gas–liquid–solid three-phase systems.

The Kyoto University team proposed, based on a two-fluid model formulation, local instant conservation equations in terms of phasic characteristic functions and a local instant interfacial area concentration. Based on this, they further derived theoretical conservation laws for two-phase flow turbulence and also for energy dissipation. Closure problems have also been discussed. The Himeji team derived an expression for the pressure gradient term with thermodynamic bases, and proposed a general formula for the virtual mass term based on the frame indifference principle. Basic equations for evaporating and condensing one-dimensional two-phase flow were derived and proposed by the JINS team with special attention paid to describing phase change terms. Of course, this work was done to improve nuclear safety analysis codes.

For three-phase flow, the Kobe University team employed a drift–flux model to obtain the basic conservation equations. In this formulation, the solid particles are treated as the additional substance affecting the gas–liquid two-phase flow structure, and their effects are considered in terms of drift velocities between the phases.

The second group includes the work being done at Tohoku University (Kamiyama and coworkers) on the basic equations of two-phase flow in a magnetic field. They considered two cases, one for magnetic fluid and the other for liquid metal. Other studies in this group of classification are proceeding at Nagoya University (Minemura and coworkers) in relation to the evaluation of the performance characteristics of a centrifugal pump and also cavitation bubble behavior. The latter problem has also been under study at Tohoku Univerity (Oba and coworkers).

3. FLOW REGIME IDENTIFICATION, MODELING AND DYNAMICS

The flow regime is, as has been well-accepted, one of the most basic and practically important characteristics of two-phase flow. The flow regimes in vertically upward co-current and horizontal two-phase flows in a simple geometry have been the subject of long-term research in Japan, as in other countries. However, recent researches on two-phase flow regimes cover various geometrical configurations of the system because of expanding demands in two-phase flow applications in various industrial areas. Flow regime means a flow classification made on the basis of the somewhat more detailed characteristic behavior of two-phase flow compared with the two-phase flow pattern.

For example, bubbly two-phase flow has such flow regimes as the wall–void-peaking regime, the intermediate-void-peaking regime and the core–void-peaking regime. However, these classifications are difficult in some practical cases, and so in this section the difference between the terms flow pattern and regime is ignored.

Researches on this subject include:

(1) the development of experimental methods and techniques to identify two-phase flow regimes;
(2) the interpretation of transition mechanisms from one regime to another and modeling for dynamic analyses;
(3) flow regime observation and/or modeling in various geometries or fluids;
(4) flow regime observation, modeling and development of transition criteria regarding nuclear thermal-hydraulics and safety; and
(5) flow regime controls.

Experimental two-phase flow regime identification methods were pursued and developed at Tsukuba University (Matsui and coworkers) for vertical, horizontal and inclined channels using the probability density function (PDF) of the pressure difference between two points. A similar method based on the differing behavior of the PDF in density (and velocity) fluctuation(s) measured by an electromagnetic flowmeter was developed at Kyoto University (Michiyoshi & Serizawa and coworkers) to identify two-phase flow regimes of liquid metal–gas two-phase flow in a magnetic field. Another method based on a pattern recognition technique using a computer was developed at Osaka University (Sekoguchi and coworkers). An attempt to study the two-phase flow regime using neutron radiography was started both at Kyoto University (Nishihara & Mishima and coworkers) (figure 1) and at Kobe University (Akagawa & Takenaka and coworkers). The advantage of this method is of course to visualize the phase distribution pattern, even in non-transparent liquids or channels. However, there are still some problems regarding spatial and time resolutions with this technique. At the moment, the method appears difficult to apply to three-dimensional flows.

The observation and modeling of bubbly two-phase flow and dynamics are described in section 5 because of the close link between turbulence and interfacial structures. The bubbly-to-slug flow transition has been jointly studied at Tsukuba University and Kyoto University (Morioka and coworkers). They related the bubble-to-slug flow transition to voidage wave instability. This flow pattern transition was experimentally studied also at Kogakuin University (Ueda and coworkers), together with detailed measurements of the phase distribution and bubble velocity and diameter distributions in vertical channels. The transition criterion for annular flow in horizontal flows was modeled at Kyoto University (Nishihara & Mishima and coworkers).

The transient behavior of gas slugs in a horizontal tube was theoretically and experimentally studied at Kobe University (Sakaguchi and coworkers). They analyzed slug velocity and length and impact force induced by liquid slugs.

Annular or annular–dispersed flow is the most important type of gas–liquid flow, hence work on this type of flow is proceeding at many laboratories. At Kyushu University (Fukano and coworkers) extensive work has been carried out experimentally and theoretically on several flow regimes in an annular flow pattern, classified on the basis of different interfacial wave

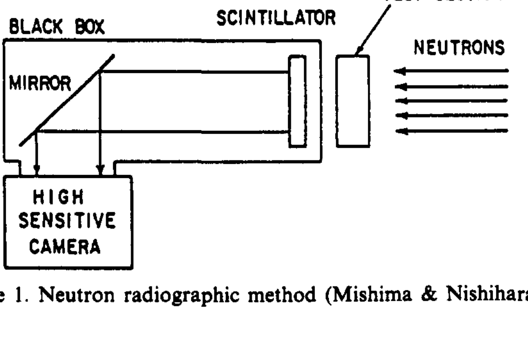


Figure 1. Neutron radiographic method (Mishima & Nishihara).

characteristics. Hold up, pressure drop and film thickness were well-correlated with these flow regimes. They further studied generation mechanisms of disturbance waves. Horizontal annular–dispersed flow in a round tube was numerically solved on the basis of the basic conservation equations at the Power Reactor and Nuclear Fuel Development Corp. (PNC) (Sugawara and coworkers). The dynamic behavior of the liquid film, wall shear stress and gas-phase turbulence in horizontal annular flow were measured and modeled at Kyoto Univerisity (Suzuki & Hagiwara and coworkers) to analyze annular flow heat transfer. Boiling and condensing annular flow behavior is being studied at Kansai University (Katsuta and coworkers) and also at Himeji Institute of Technology (Nakanishi and coworkers). Flow regimes in two-phase thermosyphons were studied at Kogakuin University (Ueda and coworkers). They examined the performance limit in terms of condensate film flooding.

Flow regimes in various geometries and fluids are of practical interest and so many people are involved in related works at many places. A typical example is a systematic work being carried out at Kumamoto University (Sato & Sadatomi and coworkers) using various non-circular channels. At Shibaura Institute of Technology (Usui and coworkers) a vertically downward flow was studied and they also conducted measurements in curved tubes (figure 2). An interesting work was started at Kobe University (Ozawa and coworkers) on two-phase flow dynamics in an elastic tube for biological applications. Along with such studies in various geometries, mentioned above, studies of two-phase flow regimes in non-ordinary fluids are also of importance in industrial applications. Such works are being carried out at Kawasaki Heavy Industries (KHI) (Nakagawa and coworkers) with cryogenic liquids such as helium, hydrogen and nitrogen. A flow map obtained by them, shown in figure 3, for liquid hydrogen is obviously different from that for an ordinary fluid system. Non-Newtonian fluid–gas two-phase flow was studied at Kobe University (Akagawa, Fujii & Takenaka and coworkers). Their results again showed different behavior from that of ordinary fluids. Two-phase flow patterns in vertically upward sodium–potassium–nitrogen gas flow at room temperature were extensively studied at Kyoto University (Michiyoshi & Serizawa and coworkers) with and without a magnetic field. In the case of no magnetic field, no significant difference was observed between liquid metal flow and ordinary fluid flow.

In relation to nuclear reactor safety and thermal-hydraulics analyses, several industries and institutes are very interested in two-phase flow regimes and dynamics in large diameter tubes. In particular, the stratified-to-slug flow transition in large diameter horizontal pipe flows has been the subject of research at the Japan Atomic Energy Research Institute (JAERI) (Tasaka and coworkers) and Mitsubishi Heavy Industries (MHI) (Tsuge and coworkers). The JAERI team clarified the effect of pressure on the flow regime transition experimentally. The MHI team systematically measured the effect of tube diameter on the transition (figure 4). Using a PWR-simulating loop, they also measured the effect of tube diameter on flow regime transitions

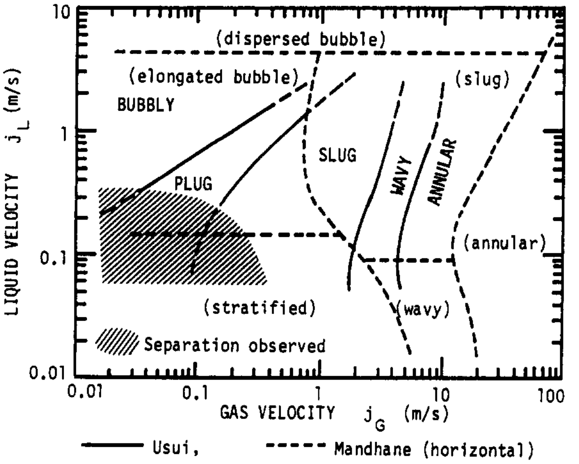


Figure 2. Flow map for air–water flow in an inverted U-bend (tube dia: 24 mm, radius of U-bend: 96 mm) (Usui).

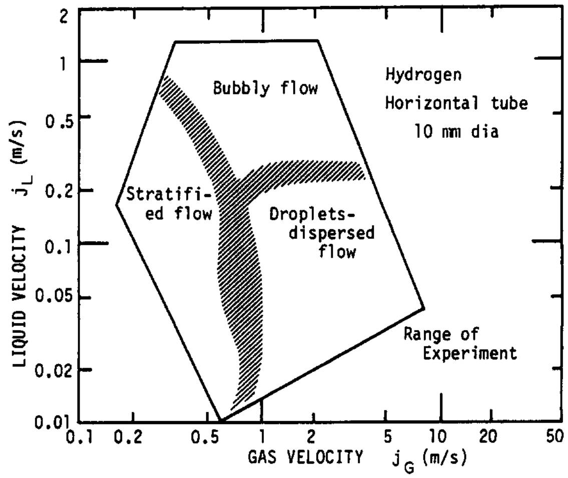


Figure 3. Flow map for horizontal liquid hydrogen two-phase flow (tube dia: 10 mm) (KHI).

and pressure drop in downward cocurrent flow. Countercurrent flow in parallel channels simulating a BWR core was studied at Hitachi Ltd (Murase and coworkers). Different flow regimes were observed at different channel locations. They developed a device to avoid the countercurrent flow limitation (CCFL).

In industrial applications, to control the two-phase flow regime under given conditions is another important problem in attaining a high efficiency of the system and in avoiding mechanical/thermal fatigue fracture of pipelines. One such attempt has been continuing at Kobe University (Sakaguchi and coworkers) to convert slug flow into bubbly flow using porous plates. They measured the spread of the bubbly flow region with different porosities, and the optimum conditions were sought. A model analysis was also made.

4. PHASE SEPARATION PHENOMENA

Phase separation phenomena are of importance in many engineering fields associated with multidimensional phenomena in two-phase flows, such as Y- or T-junctions in once-through type steam generators, phase separators, gas-purging systems, chemical plants and so on. However, it appears that little has been reported so far in Japan of these phenomena so far as the submissions are concerned. Therefore, this review includes only a limited number of works carried out at universities and in industry.

A typical systematic study of phase separation phenomena in air–water flow through symmetric and asymmetric Y-junctions has been conducted at KHI (Takemura and coworkers) to obtain some basic data for the design of a low-pressure once-through waste heat steam generator of a marine gas turbine combined cycle propulsion plant. They measured the effect of the upstream flow pattern, downstream pressure drop, inlet flow rate and the inclination of branching conduits. A model analysis was also made, but not published in an open article. At the Central Research Laboratory of the Energy Division, Mitsubishi Electric Co. (Tanaka and coworkers) phase separation was studied to obtain an optimum design of a high-performance phase separator in a vapor-compression heat pump. A schematic of their phase separator system is given in figure 5. One of the important parameters is the liquid flow ratio between the inlet flow and the outlet which is entrained in the gas. Results indicated that this parameter changed with the inlet tube location and was a function of the liquid level in the container as well as the liquid and gas phase flow rates. They also conducted flow distribution/phase separation experiments in an inclined multiple-conduit geometry and inclined Y-junction to improve the performance characteristics of a multiheat pump system. Phase separation phenomena in turbo-machinery were studied at the University of Tokyo

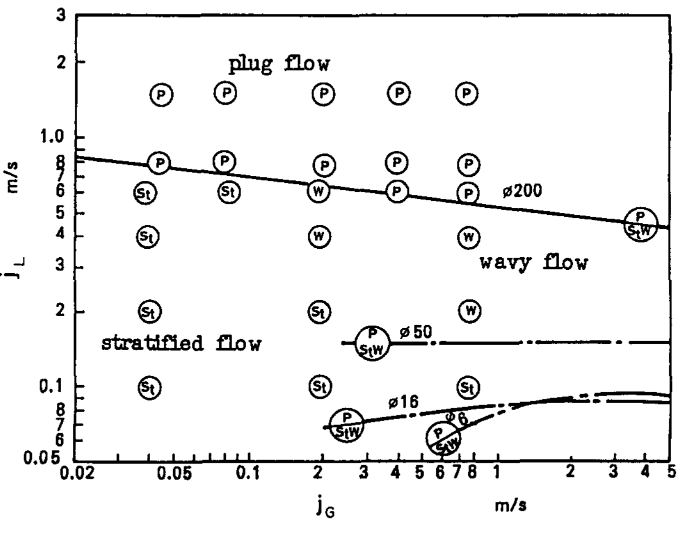


Figure 4. Flow map for horizontal air–water flow in a 200 mm dia tube (MHI).

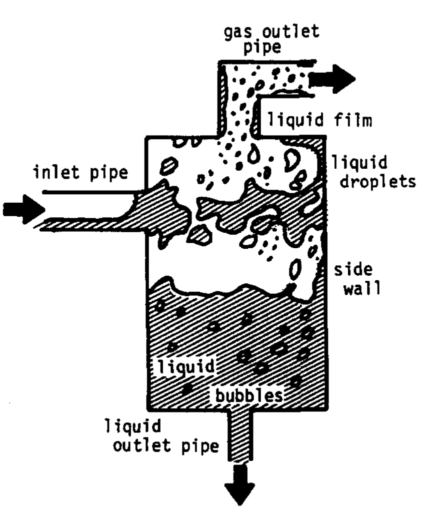


Figure 5. Conceptual design of a high-performance phase separator (ME).

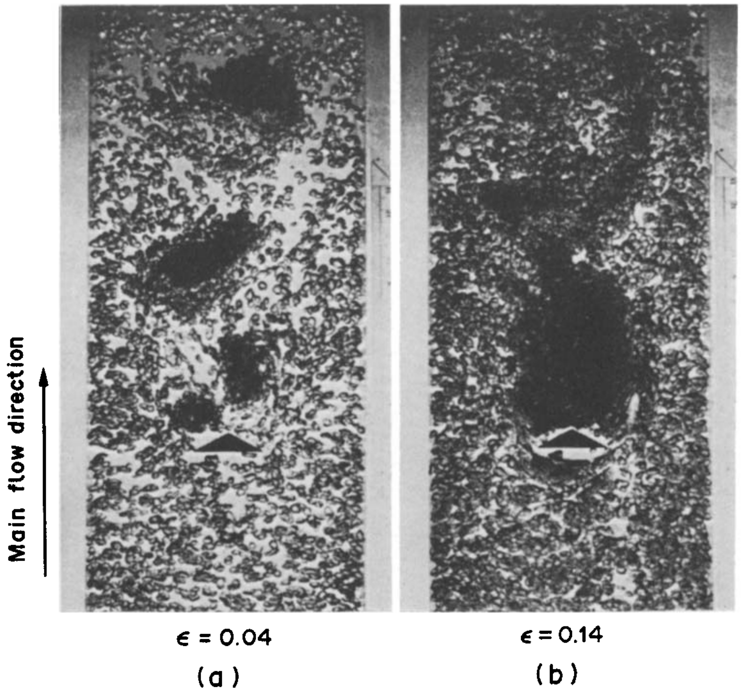


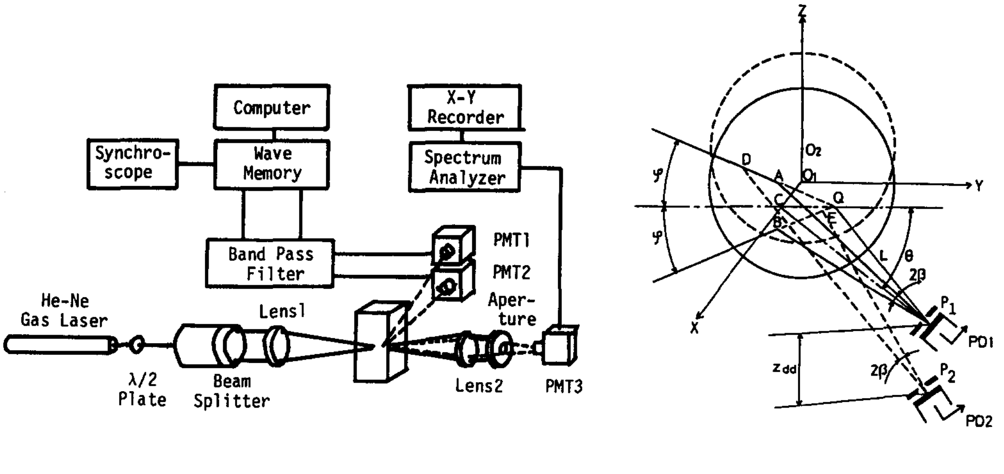
Figure 6. Phase separation around a flat plate with triangular cross section (tube dia: 30 mm, $j_L = 0.9$ m/s) (Yokosawa *et al.*).

(Ohashi & Matsumoto and coworkers) to analyze the performance characteristics of cascaded blades in an impeller. Research is underway at the Osaka Branch of the Ship Research Institute (Aya and coworkers) to understand phase separation phenomena in a blow system, using compressed air, of residual cargo in the unloading pipeline of chemical tankers which carry toxic substances.

Basic researches are also proceeding on this subject, mainly at university laboratories with rather small-scale rigs. Figure 6 shows a typical result of phase separation around or behind a plate with a triangular cross section obtained at Tokyo Institute of Technology (Inoue and coworkers). As shown in this figure, bubbles tend to collect at or near the separation point of vortices behind the body and also in the wakes. There exist two types of phase separation patterns. One is the periodic occurrence of bubble crowds like Kármán vortices in single-phase flow (left), which was observed at high liquid Reynolds number with low void fraction. The other (right) is the stagnant bubble crowd formed just behind the body—this took place at low liquid Reynolds number with high void fraction. However, it is reported that the latter was not stable and alternately changed from one to the other. They measured the shedding frequency of the Kármán vortex-like bubble crowd, and claimed that the periodic formation of bubble crowds is strongly related to the Kármán vortex. The transition mechanism from one pattern to the other is explained as an increased buoyant force acting on the large bubble crowd behind the body.

5. PHASE DISTRIBUTION PHENOMENA AND TWO-PHASE FLOW TURBULENCE

Work is proceeding in many organizations and laboratories throughout Japan on phase distribution phenomena because of the importance in many industrial applications and, also, because of it being the first step, in a sense, in a two-phase flow study. On the other hand, two-phase flow turbulence studies are rather limited to university laboratories since turbulence sounds fundamental rather than practical. Universities tend, as a whole, to devote themselves to clarifying the mechanisms or physics of phase distribution phenomena and two-phase flow turbulence. Whereas, industries are interested in more practical aspects of phase distributions in larger systems. Work done at governmental/semi-governmental institutes falls between the two. The work carried out at universities on these subjects are mostly concerned with bubbly flow. Annular/annular–



(a) Measurement system

(b) Reflection of two beams at spherical particle surface

Figure 7. Phase Doppler method (Ohba).

dispersed flow turbulence was studied at a limited number of laboratories, such as the Technical College of the University of Tokushima (Ousaka and coworkers), Kyoto University (Suzuki & Hagiwara and coworkers) etc.

As has been described elsewhere, phase distribution mechanisms in bubbly flow are closely related to continuous phase turbulence and interfacial structures. With this in mind, measurements have been conducted at the University of Tsukuba (Matsui and coworkers), Kyoto University (Michiyoshi, Serizawa & Kataoka and coworkers) and Kansai University (Ohba and coworkers) of phase distribution, turbulence and interfacial structures in round tubes or rectangular channels, paying attention to the effect of bubble size. A new LDV technique has been developed recently at Kansai University which is capable of the simultaneous measurements of local void fraction, phase velocities and bubble size (figure 7). This method uses three photomultipliers, two of them (PMT1 and PMT2) detecting the beams reflected by a bubble and the other (PMT3) detecting the beam reflected by the liquid phase. Phase distribution in various channel geometries has been systematically measured and studied at Kumamoto University (Sato & Sadatomi and coworkers).

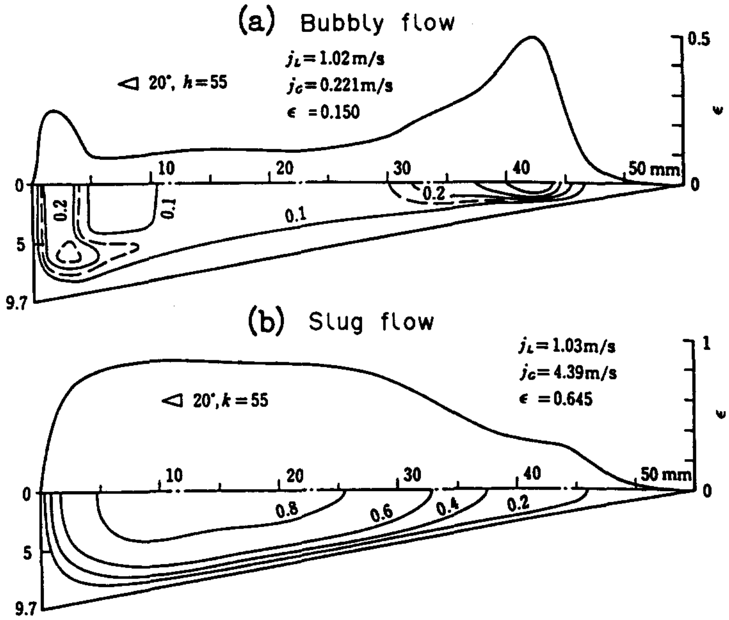
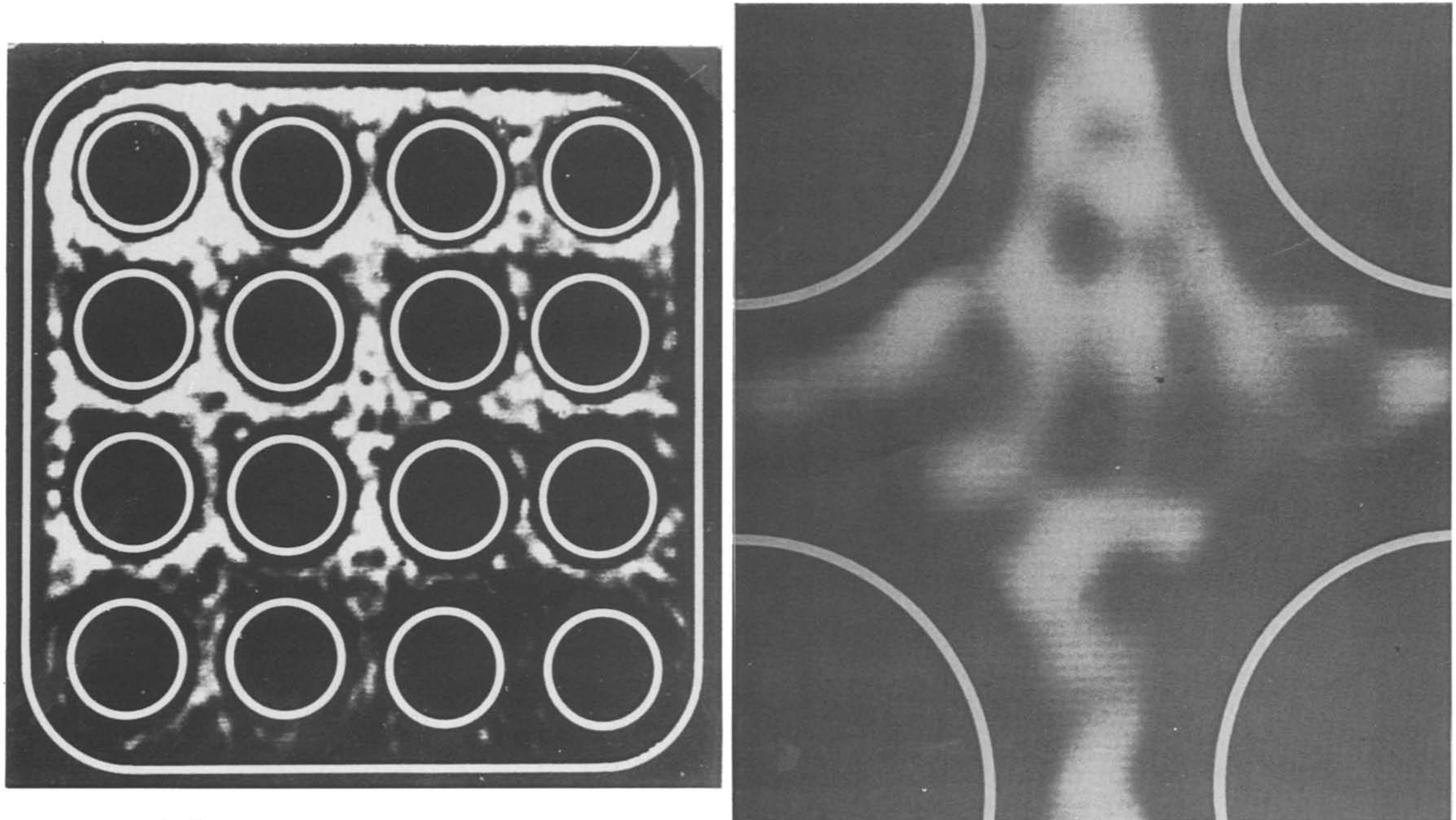


Figure 8. Equi-void-contour map for an isosceles triangular channel (Sato & Sadatomi).



(a) Whole channel cross section

(b) Center subchannel

Figure 9. Computed tomograph for phase distribution in rod bundle geometry ($w_L = 10000$ kg/h, $w_G = 3.9$ kg/h) (Toshiba).

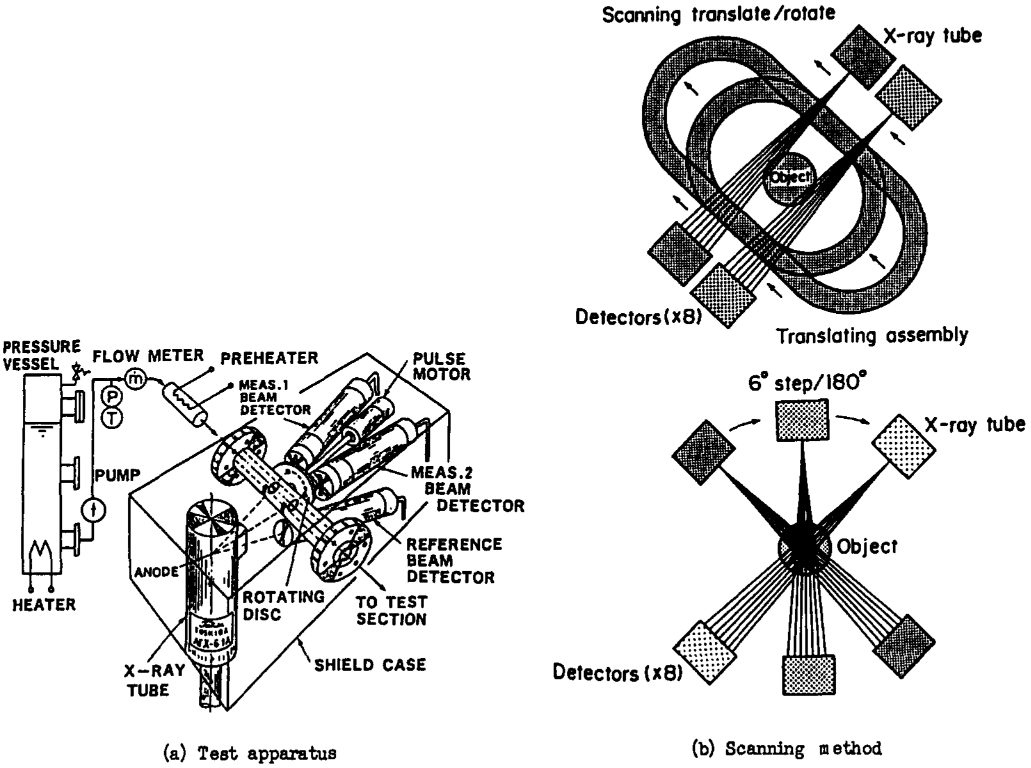


Figure 10. A dual X-ray beam scanner for transient two-phase flow measurement (Toshiba).

They tested round tubes, a triangular channel, a rectangular channel, concentric and eccentric annuli and multiple channels. Their typical results, obtained in an isosceles triangular channel, are shown in figure 8, indicating a collection of bubbles at corners. They proposed a method of calculating flow distribution and pressure drop for multiple channels. Phase distribution in annular passages was jointly studied also at Sasebo College of Technology (Furukawa and coworkers) and Osaka University (Sekoguchi and coworkers). Phase distributions in a U-bend and in a helical coil were measured at Shibaura Institute of Technology (Usui and coworkers) and Aichi Institute of Technology (Tajima and coworkers), respectively.

BWR-related works have continued at Toshiba Corp. and Nippon Atomic Industry Group (NAIG) (Mitsutake and coworkers). Phase distribution measurements in a verical 4×4 rod bundle geometry were performed at Toshiba (Iizuka and coworkers) based on a computed tomograph using an X-ray CT scanner (figure 9). Similar work was reported by Hitachi Ltd, adopting a γ-ray scanner. A technique using a dual X-ray beam scanner with a high-speed rotating disc was also developed by Toshiba (Narabayashi and coworkers) (figure 10). This method was designed to measure phase distribution and vapor velocity under high-pressure transient conditions. For accident analysis of nuclear ships, the phase distribution in an inclined simulating core containing 5×24 rods was experimentally studied at the Ship Research Institute (Iyori and coworkers) to consider the dryout margin. Phase distributions in transient boiling and in condensing flow in a vertical tube were investigated at Tohoku University (Toda & Hori and coworkers) with a computed image processing method using a high-speed videocamera.

The phase distribution in a bubble column caused by gas injection into a stagnant or near-stagnant liquid pool and bubble-induced liquid circulation are key parameters in aeration and injection metallurgy. This type of work is being carried out at the Ship Research Institute (Namie and coworkers) for application to an air bubble-type oil fence.

Concerning phase distribution mechanisms, there are several works in progress to clarify the complex phenomena by simplifying the system or focusing the point of interest. With this in mind, a lift force acting on single bubbles in a linear velocity field was studied in detail at Fukuoka University (Kariyasaki and coworkers) under laminar flow conditions. As is well-known, the phase

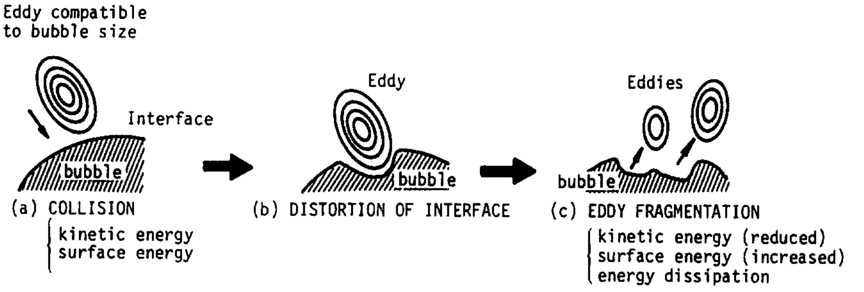


Figure 11. Turbulence eddy/interface interaction model (Serizawa & Kataoka).

distribution pattern varies significantly with bubble size and shape. The Kyoto University team (Michiyoshi, Serizawa & Kataoka and coworkers) tried to experimentally clarify the effect of bubble size on the phase distribution by using a specially designed bubble generator which is capable of varying the bubble size at given flow conditions. This work is a joint project with Ljubljana University (Zun). The effects of external forces such as gravity force, centrifugal force and magnetic force have been studied experimentally at Kobe University (Akagawa, Fujii & coworkers), at Shibaura Institute of Technology (Usui and coworkers) and Kyoto University (Michiyoshi & Serizawa and coworkers), respectively.

Finally, some theoretical works will be mentioned. As for predicting phase distribution in a round tube, bubble dispersion models were derived at Shinshu University (Hinata and coworkers) and Kyoto University (Serizawa and coworkers). A thermodynamic model was proposed by the Kobe University team (Akagawa & Takenaka and coworkers) which was based on the assumption of minimum energy dissipation. Concerning bubbly flow turbulence, a reduction in local liquid-phase turbulence has been observed and reported in some cases of bubbly flow at considerable high liquid velocity. This phenomenon is considered extremely important to the understanding of the turbulence–interfacial structure interaction mechanism, and hence to the derivation of rigorous turbulence models for two-phase flow analyses. To explain this phenomena, the Kyoto University team (Kataoka & Serizawa and coworkers) proposed a model of energy damping by bubbles which incorporates the energy exchange between turbulence kinetic energy and surface energy through an eddy fragmentation process (figure 11). The model derivation was based on a theoretical background.

6. INTERFACIAL PHENOMENA

Interfacial phenomena control mass, momentum and energy transports at the interface between phases. Many of the constitutive equations are actually related to interfacial phenomena. This subject includes two different aspects of interfacial phenomena: the first is an averaged (with respect

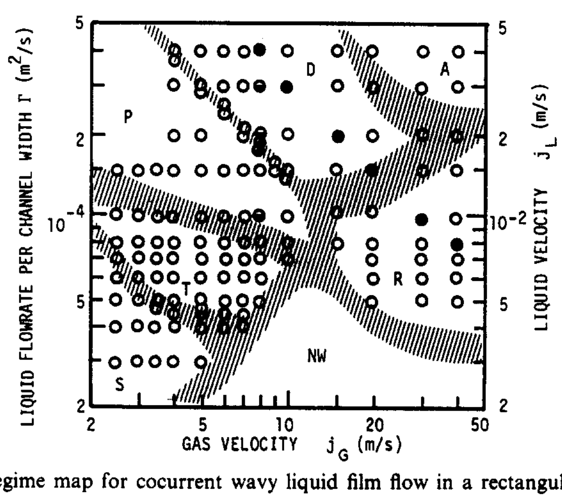


Figure 12. Flow regime map for cocurrent wavy liquid film flow in a rectangular duct (Fukano).

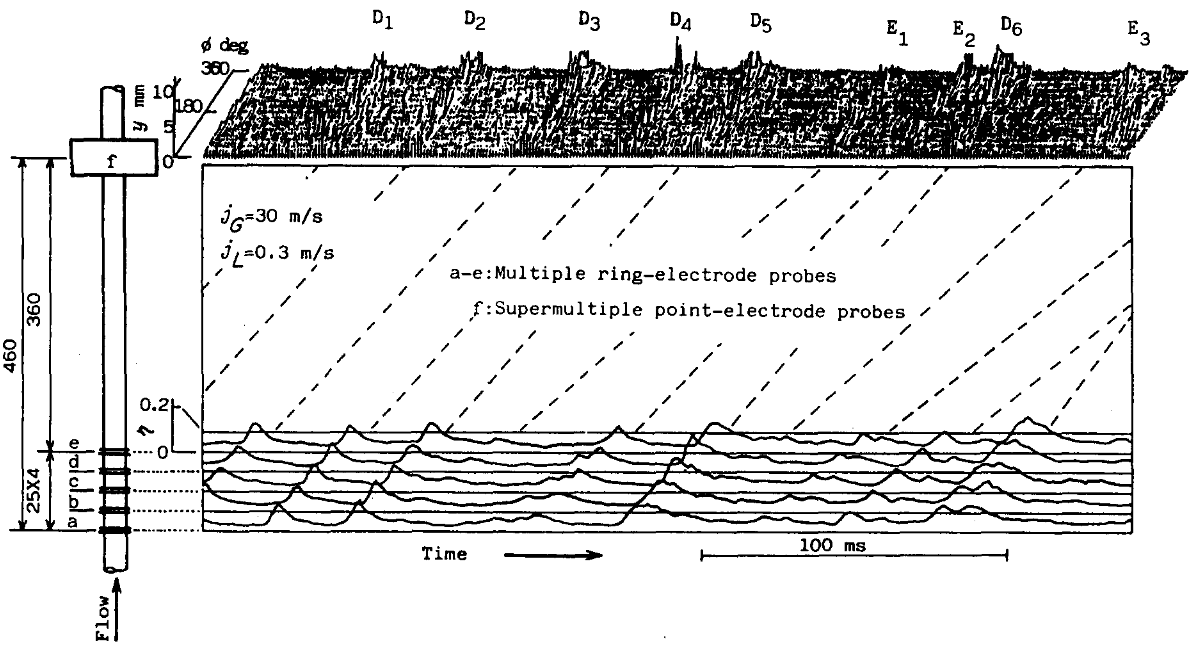


Figure 13. Interfacial contours in time sequence of three-dimensional wave height: D_i, disturbance wave; E_j, ephemeral large wave; η, cross-sectional mean liquid holdup (Sekoguchi).

to time, space or statistics) quantity such as interfacial area concentration, which is one of the most important parameters in a two-fluid model formulation of the conservation equations of two-phase flow; the second is concerned with dynamic structure or behavior of the interfaces and interfacial configuration.

As for the interfacial area concentration, measurements and analysis were carried out at Kyoto University (Kataoka & Serizawa and coworkers). They developed a new method using a probe technique to measure the local interfacial area concentration based on consideration of the local formulation and statistical averaging of the basic equations. More recently, they further proposed and developed a cross-correlation method with wider applicability, based on new concepts. Their experimental results were correlated by empirical equations, showing a power-law relation between the local interfacial area concentration and the local void fraction for bubbly flows.

Most of the work concerning interfacial phenomena paid attention to the second aspect, i.e. the interfacial structure of liquid film flow in stratified or annular/annular–dispersed flows. A typical example of this is an extensive work carried out at Kyushu University (Fukano and coworkers) which covered liquid film formation in horizontal and near-horizontal annular two-phase flows, the generation mechanism of disturbance waves in upward and downward annular flows, film breakdown and the effect of flow obstruction on liquid film flow. Figure 12 shows a flow regime map obtained by Fukano and coworkers for wavy liquid film flowing cocurrently with a gas flow in a horizontal rectangular duct. The symbols S, T, P, R, D and NW in the figure denote smooth interface, two-dimensional waves, pebble waves, ripples, disturbance waves and non-wetting region, respectively. In the non-wetting region at low liquid flow rates, they observed the breakdown of a thin liquid film. At low gas velocities, this breakdown was triggered by the formation of a viscous wave on the liquid film. On the other hand, when the gas velocity was sufficiently high, the disturbance waves prevented the film flow from being broken up. They also claimed that the disturbance wave was formed at interfacial fluid particle velocities exceeding the wave velocity.

An attempt to visualize multidimensional (more exactly, time sequential) aspects of liquid film flow with different types of surface waves was successfully tried at Osaka University (Sekoguchi and coworkers) using a specially fabricated multisensor probe and a signal processor. An example of their results is presented in figure 13, indicating the formation of ephemeral large waves as well as disturbance waves. Based on this kind of observation, they pointed out the existence of a huge wave flow regime for the transition to annular flow in a vertical channel.

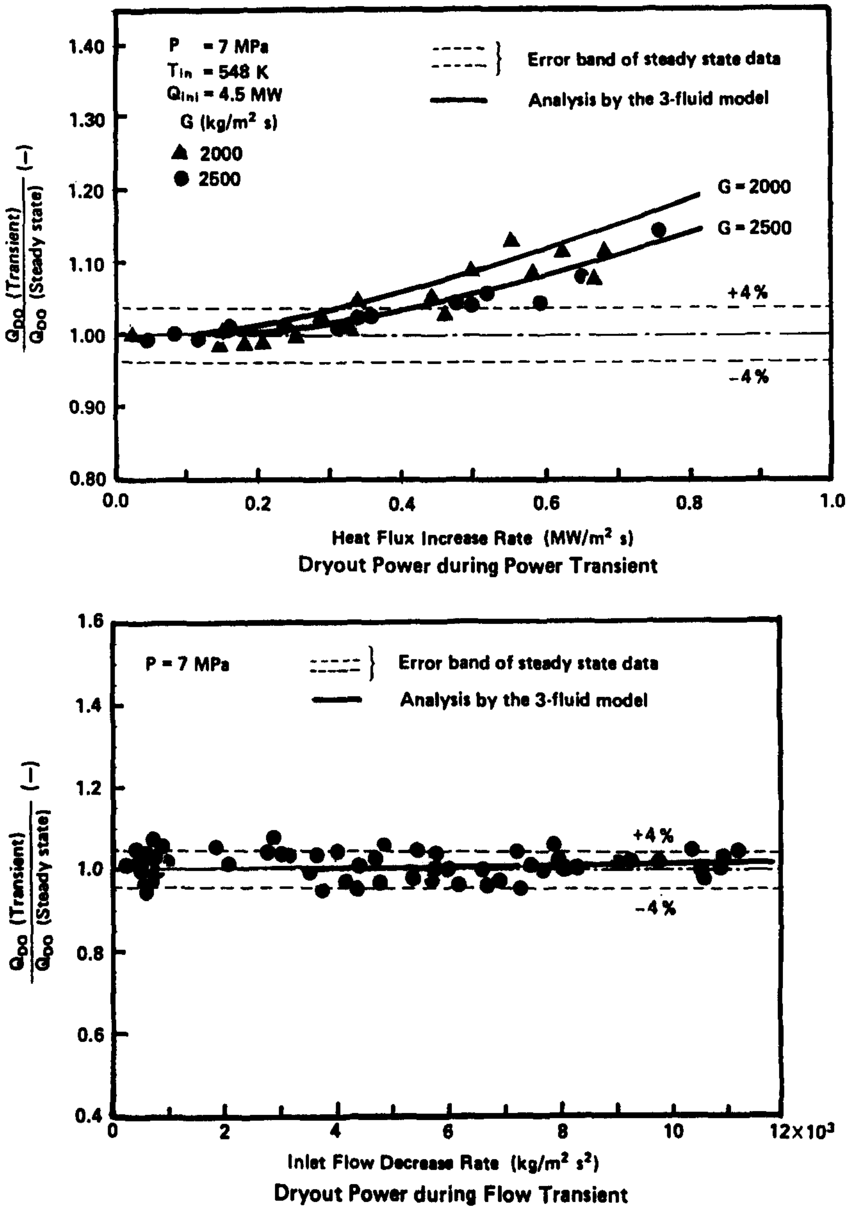


Figure 14. Dryout power in flow and power transients (Sugawara—PNC).

The dynamic behavior of the film flow and surface waves as well as droplet entrainment under adiabatic conditions were experimentally studied and, in some cases, visualized at Kansai University (Ohba and coworkers) using a fiber optic liquid film sensor, and at Kyoto University (Mishima & Nishihara and coworkers; Suzuki & Hagiwara and coworkers). Most of these studies were discussed in relation to the transition mechanism to annular flow. Characteristics of disturbance waves in boiling two-phase flow at relatively high pressure were studied at Takamatsu College of Technology (Yamauchi & Sawai and coworkers) to clarify the dryout mechanism. The results indicated that disturbance waves had a close relation with wall temperature fluctuation at near dryout. A non-equilibrium condition was suggested between base film flow and disturbance waves.

7. NON-EQUILIBRIUM MODELING

To analyze transient behavior in boiling and condensing two-phase flows, non-equilibrium modeling is essential. This is particularly important in nuclear reactor accident analyses. In boiling

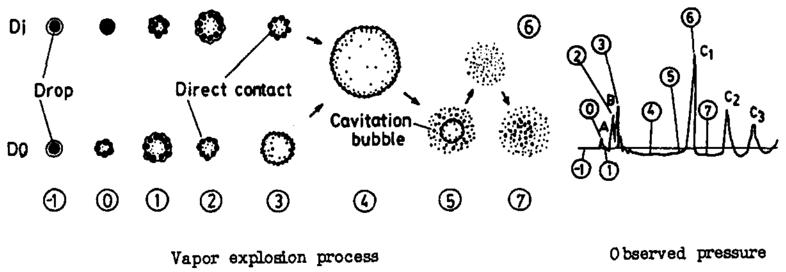


Figure 15. A model of the vapor explosion mechanism in the two-droplet case (Iida).

two-phase flow analysis, power excursion, flow reduction and depressurization are basic frames of non-equilibrium modeling.

At PNC (Sugawara and coworkers), a series of measurements were carried out of dryout heat flux in both power and flow transients. As shown in figure 14, their results indicated, in both cases, higher transient dryout heat flux than in the steady state. To predict these experimentally observed trends, they conducted a numerical analysis based on the two-fluid three field model and obtained satisfactory results. A flashing flow under rapid depressurization in a horizontal tube was investigated at Ibaraki University (Kaminaga and coworkers). They measured the relaxation time for the thermal equilibrium state. The bubble growth in depressurization was studied at Tohoku University (Toda and coworkers). They claimed that, because of the evaporation rate at the vapor–liquid interface enhanced by decompression waves, the bubble grew more rapidly during that transient than predicted by theory. They also studied condensing two-phase flow. Computer code development will be summarized later.

8. VAPOR EXPLOSION

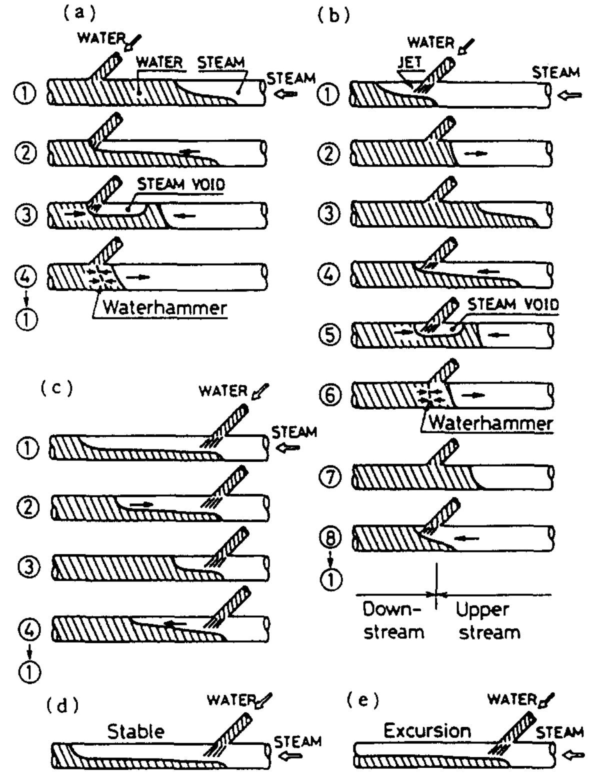
Vapor explosion studies are needed in relation to nuclear reactor safety problems because of the vapor explosion-induced pressure pulses which occur after melted fuel–coolant interactions. However, only a small number of works were sent to the present authors on this subject. All of these works are basic studies at universities aimed at clarifying the vapor explosion mechanism.

The vapor explosion produced by single and plural liquid droplets of molten $LiNO_3$ dropped in liquid ethanol has been studied at Yokohama National University (Iida and coworkers). They found that the vapor explosion took place when the droplet came into contact with triggering needles and that it was triggered also by pressure pulses originating from the vapor explosion of other droplets. This model is illustrated schematically in figure 15 for the two-droplet case. A similar experiment was conducted at the University of Tokyo (Shoji and coworkers), using a molten tin droplet into stagnant water. In this experiment, the emphases were on the effects of the initial temperatures of water and tin, the system pressure, the water level and the temperature distribution in water. An analytical model similar to that of Inoue *et al.* (1981) was proposed. In their work, observations were also made with water flowing over a stationary molten tin layer. Tokyo Institute of Technology (Inoue and coworkers) continued their work on this subject.

9. THERMAL-HYDRAULIC INSTABILITIES

Thermally-induced flow instabilities have been studied for safety consideration of nuclear reactors and for the design of natural circulation boiler, cryogenic evaporators, thermosyphons and so on.

BWR-related work was actively pursued in industry, e.g. a joint research by several electric power suppliers, Hitachi Ltd and Toshiba, using BWR-simulating cores and bypass channels (refer to illustrative papers in TOJ-4 and TOJ-5 in the appendix). Experimental conditions were mostly chosen so as to cover the BWR operation. The instability criterion or threshold power for density wave oscillation and the critical power at boiling transition with density wave oscillation were sought. Regarding PWR operations, flow instabilities in a large-diameter inverted U-tube (6.6 m long 200 mm i.d. with 750 mm bend radius) were studied at MHI (Tsuge and coworkers). Flow



(a) On-off mode with non-periodic occurrence of water hammer
(b) On-off mode with periodic occurrence of water hammer
(c) Flow oscillation without water hammer
(d) Stabe liquid plug
(e) Retreating liquid plug

Figure 16. Oscillation modes and interfacial configuration in a horizontal channel simulating a PWR cold leg (Nariai & Aya).

patterns downstream of the bend were observed and the period of the instability was discussed. Pressure and flow oscillations in a cold leg during a postulated loss-of-coolant accident (LOCA) were pursued at the Ship Research Institute (Aya and coworkers) jointly with the University of Tsukuba (Nariai and coworkers). Based on their observations, they classified the flow oscillation modes in the cold leg as on–off oscillation, plug oscillation, stable mode and excursion mode, as shown in figure 16. The oscillation limit and frequency of the oscillations were calculated by a linear analysis.

Density wave oscillation in a natural circulation steam generator loop was analyzed by linear stability analysis using the D-partition method at Himeji Institute of Technology (Nakanishi and coworkers) jointly with Takamatsu College of Technology (Yamauchi and coworkers). A similar problem was treated at Hitachi Zosen Ltd (Furudera and coworkers) using the Nyquist method. Work has been done on flow instability in parallel channels at Kobe University (Akagawa, Fujii & Takenaka and coworkers) with upward liquid nitrogen-vapor flow, and at Kyushu University (Hasegawa & Fukuda and coworkers) with downward freon-113–vapor flow. In the former case, they controlled the boiling mode at the inlet section, and, thus, the experiment was conducted under conditions covering the range from bubbly to annular flows (in the nucleate boiling case) and also the range from inverted annular to dispersed flows (in the film boiling case). Linear stability analysis together with the drift–flux model is reported to have a well-correlated measured stability criterion in both cases. Flow instabilities in vertical and horizontal nitrogen evaporators were pursued experimentally and analytically by the Kyushu University team also.

Studies of the dynamic behavior of a thermosyphon have been conducted at Kyushu University (Fukano and coworkers; Hasegawa & Fukuda and coworkers) and Kogakuin University (Ueda and coworkers). Excursion and oscillation of the heated wall temperature took place when the heat

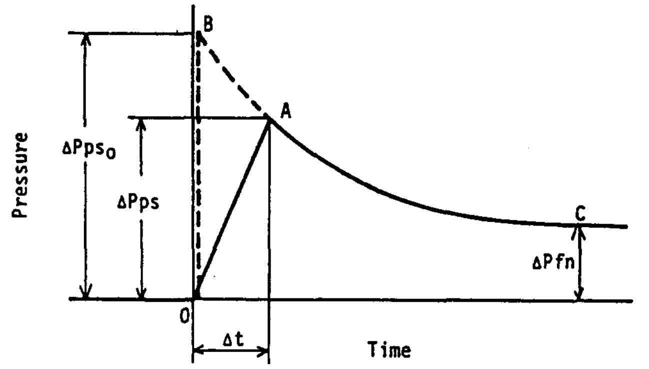


Figure 17. Pressure transient model (Akagawa, Fujii & Takenaka).

input exceeded a certain critical value. Empirical correlations were proposed by them which consider the effects of entrainment and flooding.

10. PRESSURE AND VOID WAVE PROPAGATION AND SHOCK PHENOMENA

Shock phenomena and pressure and void wave propagation in two-phase flow are the important subject of research associated with safety operations and designs of piping systems of nuclear power plants, boilers, chemical plants etc. These phenomena are usually caused by the rapid closure or opening of a valve, water hammer, cavitation, rapid boiling of a liquid (like vapor explosion and voiding), bubble collapse and high-speed two-phase jets. Therefore, most of the work is specifically oriented to particular systems.

Nuclear reactor-oriented work was performed at Hitachi Ltd (Kawasaki and coworkers) and also at the Ship Research Institute (Aya and coworkers) in conjunction with the University of Tsukuba (Nariai and coworkers). The former research dealt with supersonic two-phase jet impinging on the wall for consideration of pipe rupture accidents in nuclear reactors. A three-dimensional homogeneous equilibrium model was adopted to analyze the results. In the latter work, the interest was in the water hammer phenomena taking place in a cold leg with flow oscillations. They found that the water hammer was caused when the steam plug collapsed during flow oscillations. Cavitation characteristics in an FBR inducer pump were the research topic at KHI (Roko and coworkers).

Shock phenomena caused by the rapid closure of a valve and the subsequent pressure propagation were studied in detail at Kobe University (Akagawa & Fujii and coworkers) both experimentally and analytically. Their experiment was conducted in a horizontal tube with refrigerant R-113 for void fractions ranging from 0 to 0.22. This work is a long-term research project at Kobe University. Experimental results indicated that both the pressure surge and the propagation velocity decreased with increasing void fraction. They proposed a mechanistic model based on one-dimensional homogeneous two-phase flow formulations to calculate the pressure transient, as illustrated in figure 17. They numerically solved a set of equations to obtain the potential surge ΔP_{ps}, the time delay Δt, the equilibrium pressure rise ΔP_{fn} and the propagation velocity of the pressure wave. They obtained analytical solutions also. These calculated results were satisfactorily compared with experiments.

The magnetohydrodynamic effect on pressure pulsation at near critical flow conditions was studied analytically at Tohoku University (Kamiyama and coworkers) for both liquid metal–gas and magnetic fluid–gas two-phase bubbly flows in a converging–diverging duct. The effect of applying a magnetic field was reported to suppress pressure pulsations and to shift the initiation point of the critical flow condition down the throat of the nozzle.

Basic studies of cavitation phenomena were made independently by two Tohoku University teams (Kamiyama and coworkers; Oba and coworkers). The Kamiyama team conducted experiments on cavitation at a square-edged orifice in benzene, kerosene, gasoline and freon-12. The results were compared with predictions based on an analogy to two-phase critical flow. Very good observations and an extensive results were reported by the Oba team. Studies were conducted also at the University of Tokyo (Ohashi and coworkers) in relation to two-phase pump

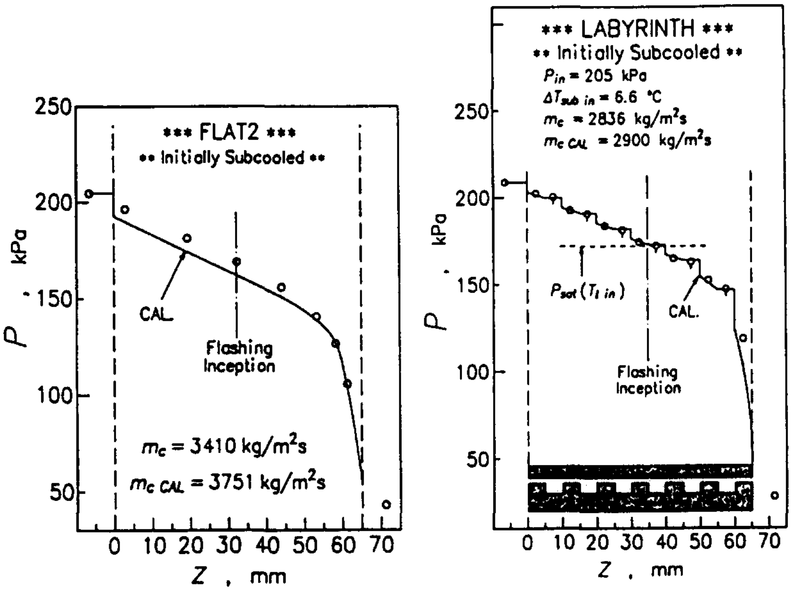


Figure 18. Experimental and calculated pressure distribution in a narrow rectangular channel and a labyrinth (Hijikata).

performance characteristics. The internal phenomena of a bubble and the non-linear behavior of acoustic cavitation are also being studied at the University of Tokyo (Matsumoto and coworkers).

Pressure pulsations induced by the rapid growth and collapse of a bubble during transient boiling in a narrow channel were studied at Tokyo Institute of Technology (Kozawa and coworkers). A bubble growth and collapse model was proposed and solved using homogeneous two-phase flow equations. Also studied at Tokyo Institute of Technology (Nakagawa and coworkers) was the transient behavior of a bubbly cavity train.

Voidage wave propagation and instability problems were dealt with at the University of Tsukuba (Morioka and coworkers—Morioka moved recently to Kyoto University) to explain the transition from bubbly to slug flow in a vertical channel.

11. CRITICAL TWO-PHASE FLOW

Critical flow with two-phase flashing has been studied extensively because of the importance of this phenomenon in the analysis of hypothetical accident sequences in nuclear reactor systems and other industrial applications.

Most previous studies on critical two-phase flow are related to flow through relatively large-diameter tubes and orifices. However, the importance of studying flows of flashing liquid and vapor through narrow cracks and nozzles has been acknowledged. A typical example of this type of work was carried out at Tokyo Institute of Technology (Hijikata and coworkers). In their experiment, flashing flow of initially subcooled water through a narrow rectangular channel with 0.125 or 0.170 mm width and a labyrinth was tested by visual observation and by measuring flow parameters for various inlet conditions. A theoretical model was also proposed which considered the effect of thermal non-equilibrium between the phases. This model stressed its ability to accurately predict the pressure distribution in the channel. Some of their results are given in figure 18.

Flashing critical flow and two-phase jets through short cylindrical nozzles were studied at Tokyo Institute of Technology (Inoue and coworkers). The results indicated that the effect of thermal non-equilibrium reduced with a decrease in the length-to-diameter ratio of the nozzle. They also measured reaction force and impact pressure. At PNC (Mochizuki and coworkers), the discharge flow rates from circular holes (2–40 mm dia) and slits with various aspect ratios were measured as a function of pressure (2–7 MPa) and subcooling (0–264 K).

Kobe University (Akagawa & Fujii and coworkers) has an experimental facility to investigate the basic performance characteristics of a converging–diverging nozzle of a turbo-type total flow

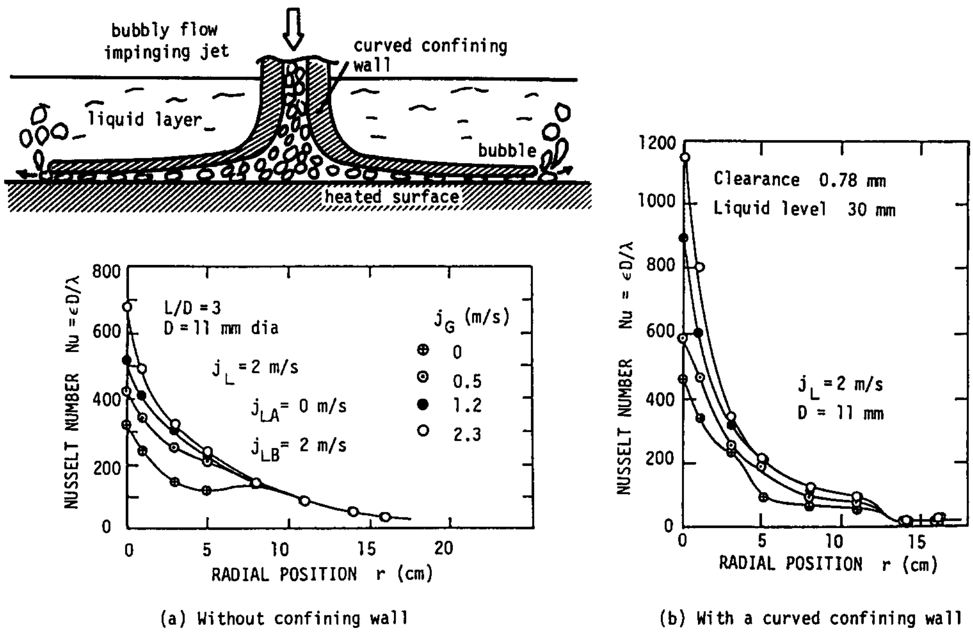


Figure 19. Heat transfer augmentation by an air–water bubbly flow impinging jet with confining walls (Serizawa & Michiyoshi).

turbine where initially subcooled water is expanded by flashing. This study originated from an intention to achieve a higher efficiency of the total energy utility of plants or systems using water-dominated geothermal resources and waste heat from factories by improving turbine efficiency.

12. FORCED CONVECTIVE HEAT TRANSFER—HEAT TRANSFER ENHANCEMENT

There are practical requirements to develop new methods of heat removal with high efficiency in designing advanced heat transfer equipment. In view of this, a number of research laboratories are working on this subject. A variety of efforts are being made in industrial applications and also with university laboratory setups: e.g. the helical coil evaporator at Aichi Institute of Technology (Watanabe and coworkers); the heat transfer augmentation by horizontal tube arrays at Kyushu University (Fujita and coworkers); the horizontal evaporator tubes with internal spiral grooves at Kyushu University (Yoshida and coworkers); the surpentine-tube heat exchangers at Waseda University (Katsuta and coworkers); the heat transfer enhancement by a bubbly flow impinging jet or a binary mixture impinging jet with confining walls at Kyoto University (Serizawa and

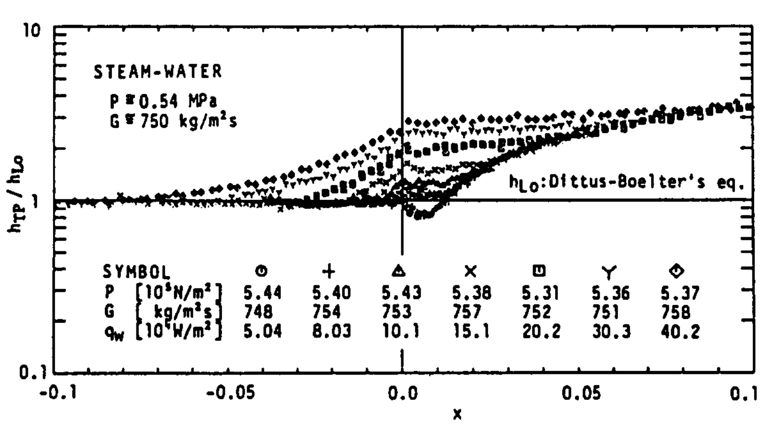


Figure 20. Forced convective heat transfer in a vertical channel (Sekoguchi & Kaji).

coworkers) (figure 19) and by water droplets suspended in a gas stream at Tohoku University (Aihara and coworkers); and mist or spray cooling for LOCA analysis at Tohoku University (Toda and coworkers), the University of Tokyo (Shoji and coworkers), the Ship Research Institute (Namie and coworkers) and Toshiba Co. (Kuwako and coworkers); and so on.

The use of non-azeotropic binary mixtures as the working fluid is one way of achieving effective energy utilization. At Tokyo Institute of Technology (Hijikata and coworkers) an attempt has been made, using a mixture of two immiscible liquids, to improve the thermal efficiency of the Rankine cycle which utilizes a small temperature difference. The boiling heat transfer characteristics of non-azeotropic binary mixtures were also studied at the University of Tokyo (Saito & Hihara and coworkers) using both R12–R22 and R114–R22 mixtures flowing in a uniformly heated horizontal tube, at Kyushu University (Fujita and coworkers) using an R11–R113 mixture in a vertical tube, at Himeji Institute of Technology (Nakanishi and coworkers) and at Toshiba Co. (Hashizume and coworkers) with regard to high efficiency heat pumps.

Recently, cryogenic fluids, such as liquid helium and hydrogen, have become more important in practical applications. Among the submissions from many organizations, however, only work at KHI (Roko and coworkers) was found on the heat transfer characteristics of such fluids. Increasing demands for developments in fusion technology and liquid metal MHD generators have promoted some basic researches on the heat transfer characteristics in liquid metal–gas two-phase flows with and without a magnetic field. This type of research is being continued at Tokyo Institute of Technology (Inoue & Aritomi and coworkers) with lithium–helium stratified and annular–dispersed flows in a horizontal rectangular duct, at Kyoto University (Michiyoshi & Serizawa and coworkers) with vertically upward sodium–potassium–nitrogen flow in a round tube covering bubbly, slug, churn and annular–dispersed flows, at Tohoku University (Toda and coworkers) with sodium mist flow and at Osaka University (Miyazaki and coworkers) with lithium–gas flows.

More fundamental studies on heat transfer in two-phase flow using water were made at Osaka University (Sekoguchi and coworkers) and Kyoto University (Suzuki & Hagiwara and coworkers) for vertical tubes, at Kyoto University (Mishima & Nishihara and coworkers) for upward and downward flows in a vertical rectangular channel, at Kansai University (Katsuta and coworkers) for inclined tubes and at Tokyo Institute of Technology (Hijikata and coworkers) for counter-current flow. Figure 20 shows a typical result obtained very carefully by Sekoguchi and coworkers. The variation of the heat transfer coefficient with the two-phase flow pattern is very clearly seen in this figure. All of their experimental heat transfer data for forced convection were well-correlated by a Bennett-type empirical correlation. The work of Hijikata and coworkers included a theoretical analysis based on a low Reynolds number $k - \epsilon$ turbulence model. Continuing efforts are being made at Kyoto University (Suzuki & Hagiwara and coworkers) to develop methods of numerical analysis for predicting the heat transfer characteristics of forced-convection filmwise condensation

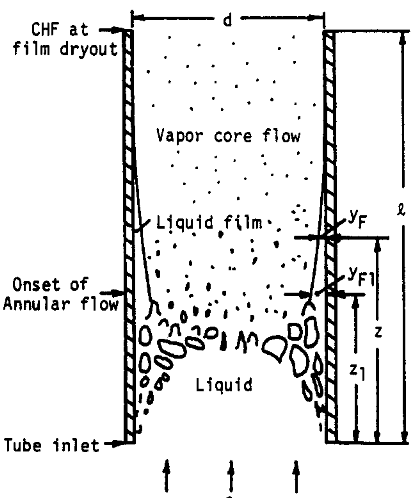


Figure 21. A model of liquid film behavior at CHF (Katto).

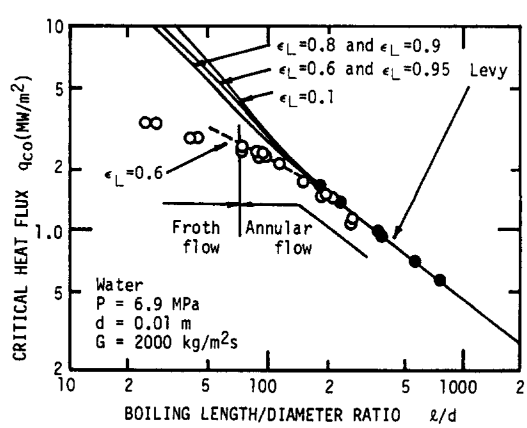


Figure 22. CHF vs length-to-diameter ratio for saturated flow (Katto).

as well as those of the entrance region of vertically downward two-component two-phase annular flow.

13. FORCED CONVECTIVE HEAT TRANSFER (CHF)

A number of research teams have been devoted to studying CHF because of its importance in many practical and industrial applications. Most of the work is, however, rather fundamental. Among them, the most comprehensive work is that underway at University of Tokyo (Katto and coworkers—presently at Nihon University) covering CHF both for internal and external flows over a wide range of parameters. They produced improved versions of previously proposed generalized correlations, and presented more insight into flow boiling CHF. Figure 21 illustrates the liquid film behavior at CHF in vertical annular flow proposed by the Katto team. Their model assumed dryout of the liquid film at critical conditions based on Levy's annular flow model. A comparison between the prediction and the experiment is shown in figure 22, where the solid lines denote the prediction and the circles denote the experimental data. The agreement appears good for length-to-diameter ratios $l/d > 200$. However, the prediction in the range $l/d < 200$ shows noticeable deviation from the trend in the experiment, though they are still in annular flow. To explain this, the critical liquid film concept was introduced. The result based on this new model is also shown by the dashed line in figure 22.

CHF at nearly atmospheric pressure and low velocity conditions was measured at Kyoto University (Mishima & Nishihara and coworkers), including the effects of channel geometry and flow instability. Their result for channels with unheated walls indicated lower values than the prediction by the Katto correlations. Bubble behavior near the heated wall was studied at Kogakuin University (Ueda and coworkers) to understand the process of wall temperature excursion at subcooled flow boiling CHF. Their conclusion was that the large wall temperature rise associated with CHF was caused by periodical overheating of the wall due to the passage of coalesced bubbles.

CHF for R-113 in narrow channels at high pressure (1.1–2.1 MPa), high mass flux (10,000–20,000 kg/m^2 s) and high subcooling (20–80 K) was studied at the University of Tokyo (Hirata and coworkers). Other work on CHF in boiling freon is being carried out at Kyushu University (Yoshida and coworkers) at subcritical pressures in tubes and at Osaka University (Kaji and coworkers) in helically coiled tubes.

In relation to the design limit of a BWR, the boiling transition or critical power was studied under a joint research program between several electric power supplier and nuclear industries (see illustrative papers in TOJ-4 in the appendix). Some of the recent studies on CHF oriented to nuclear reactor thermal-hydraulics placed emphasis on the tight lattice core of high conversion light water reactors (HCLWR), e.g. the work at Tokyo Institute of Technology (Inoue and coworkers) and NAIG (Arai and coworkers). At the Ship Research Institute (Iyori and coworkers) work is underway to clarify the dryout conditions in a nuclear merchant ship under severe accident conditions such as a grounding. A full-scale dryout test of an advanced thermal reactor (ATR) during flow and power transients was simulated at PNC (Sugawara and coworkers).

14. POST-DRYOUT HEAT TRANSFER

The majority of the work on post-dryout heat transfer is related to the heat removal during a LOCA in a water reactor. The extensive research programs being pursued at JAERI include post-dryout heat transfer experiments at high pressures (Kumamaru and coworkers), study of PWR reflood phenomena using the large-scale cylindrical core test facility (CCTF) (figure 23) (Murao and coworkers) and the slab core test facility (SCTF) (figure 24) (Murao and coworkers). It was revealed from the 25-rod bundle experiment at pressures from 3 to 12 MPa (Kumamaru and coworkers) that the post-dryout heat transfer strongly depends upon the wall superheat. The heat transfer coefficient was high in the low wall superheat region and it increased with increasing quality and pressure. To account for this trend, new correlations have been proposed by improving the Groeneveld correlation. A large amount of data accumulated from these research projects have

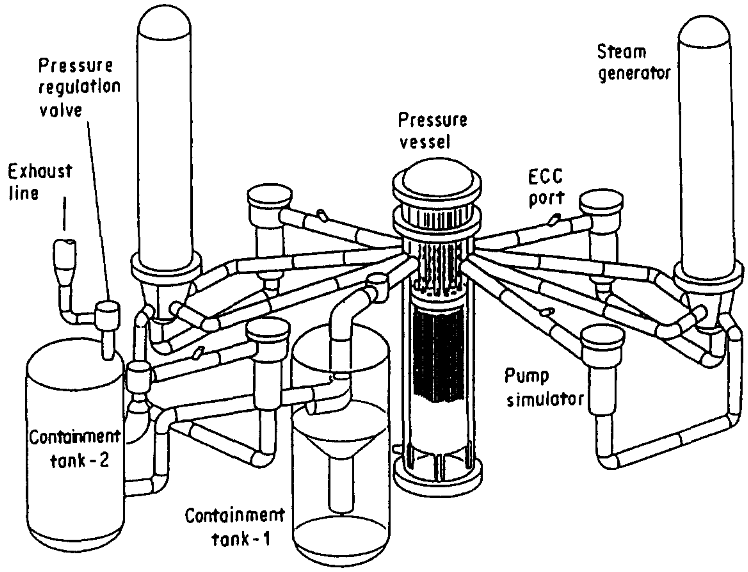


Figure 23. A schematic of the CCTF (JAERI).

been used to evaluate the predictability of computer codes such as TRAC and REFLA. The joint research program between the electric power suppliers and nuclear industries is pursuing post-dryout heat transfer and rewetting phenomena during a BWR LOCA also. Post-dryout behavior of an ATR rod bundle was investigated at PNC (Sugawara and coworkers).

Post-dryout heat transfer is also an important problem in many industrial applications other than nuclear reactors, e.g. high-performance once-through steam generators, steam-heating shell-and-tube LNG evaporators, two-phase thermosyphons etc. Such work is underway at Kyushu University (Yoshida and coworkers) on the post-dryout heat transfer to freon at high pressures in the subcritical region, at KHI (Roko and coworkers) for the design of a compact steam generator, at Mitsui Engineering and Shipbuilding Co. (Mori and coworkers) in relation to LNG evaporators and at Kyushu University (Fukano and coworkers) and Kumamoto University (Imura and coworkers) for thermosyphons.

Inverted annular flow has attracted a great deal of attention recently in relation to the LOCA sequence, the design of FBR steam generators and cryogenic fluid flow in a conduit. The work done at Kobe University (Akagawa, Fujii & Takenaka and coworkers) clarified the relationship between flow regime, heat transfer and pressure drop over the range from inverted annular to dispersed flow. Inverted annular flow is a current research subject, and has been for many years, at Tokyo Institute of Technology (Inoue & Aritomi and coworkers), using freon-113 in vertical and horizontal tubes with different diameters. They proposed non-dimensional empirical correlations for both the heat transfers coefficient and vapor film thickness in inverted annular flow in terms of the thermal equilibrium quality, wall superheat, tube diameter and Nusselt and Reynolds numbers, which are defined based on the tube diameter.

15. CODE DEVELOPMENT AND VERIFICATION

Computer code developments and their verification tests on the thermal-hydraulics of two-phase flow have been carried out to confirm the best estimate method for safety and operational conditions of LWR, ATR and FBR in nuclear power institutes and plant industries. Only a brief review will be made here, based on the submissions from several institutes and industries. Major computer codes developed in Japan were summarized and tabulated previously in the report of the Atomic Energy Society of Japan (Nariai 1986).

A one-dimensional transient two-phase analyzer MINCS was developed at JAERI (Akimoto & Hirano and coworkers) which can treat various two-phase flow models and can compare the availability of many constitutive equations. This code has been used widely for the evaluation and

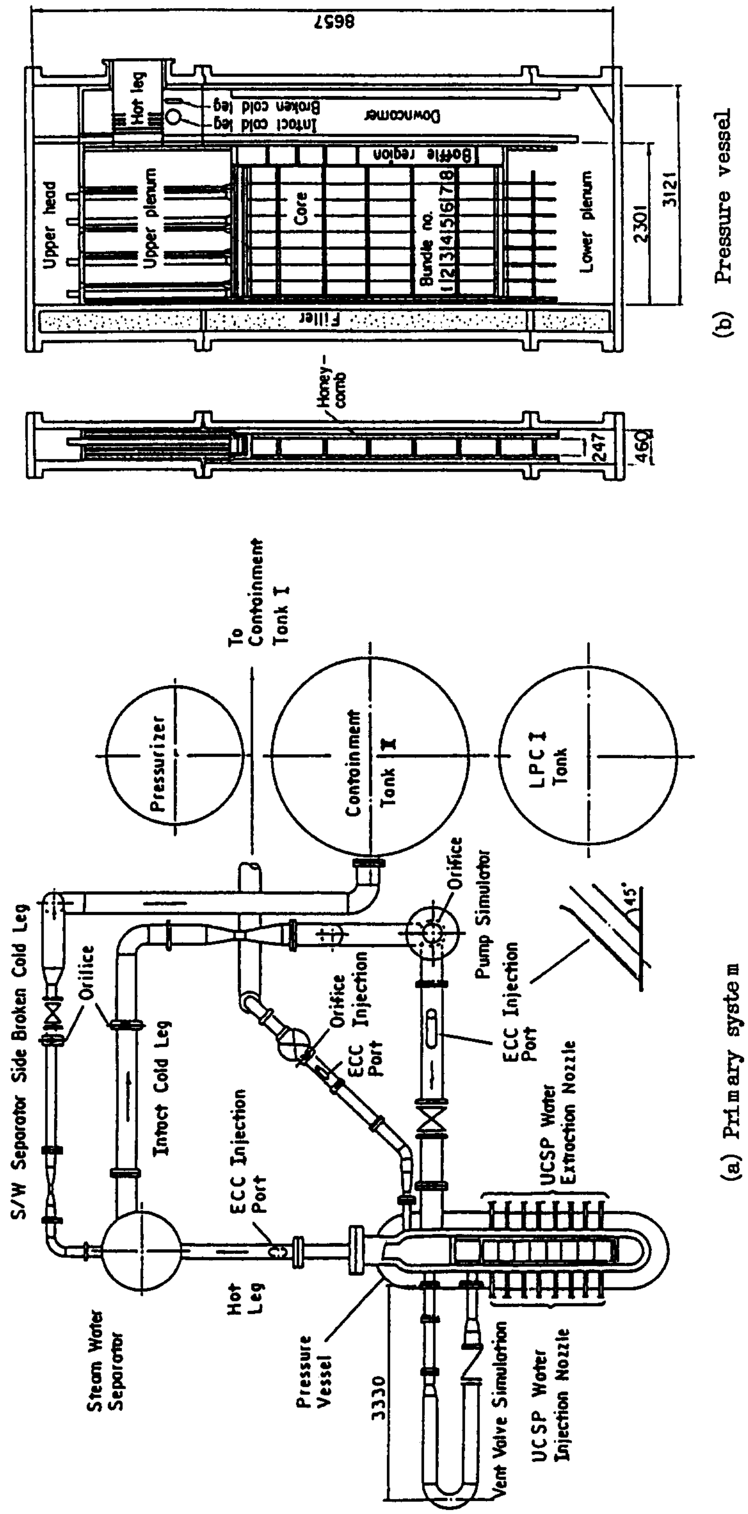


Figure 24. A schematic of the SCTF (JAERI).

development of two-phase flow analysis at JAERI, such as vessel blowdown tests and thermodynamic non-equilibrium problems, and was applied recently to benchmark problems, the results of which were presented at the *3rd International Workshop on Two-phase Flow Fundamentals*, cosponsored by U.S. DOE and EPRI, held at Rensselaer Polytechnic Institute, New York, 1986. The J-TRAC codes were developed at another JAERI laboratory (Murao & Akimoto and coworkers) to analyze core cooling during the reflood phase in a LOCA. These codes were applied to the experiments with a small-scale reflood test facility and a cylindrical core test facility (SCTF) (both at JAERI), and an upper plenum test facility (UPTF) (in F.R.G.).

A subchannel analysis code SABENA and a whole-core analysis code ARAMADA have been developed at PNC (Ninokata and coworkers) for sodium boiling analysis in FBRs, and were applied to the experiments with the 7-, 19-, 37-pin bundle loss of flow test (LOFT) and also to 37-pin bundle natural circulation boiling experiments conducted by PNC and KfK. The SABENA code was also presented at the *3rd International Workshop on Two-phase Flow Fundamentals.* A three-fluid analysis code to predict transient dryout phenomena in ATRs was proposed at PNC (Sugawara and coworkers). A similar code for BWR transients was developed at NAIG (Mitsutake and coworkers).

A 2V2T model using the volume junction method was developed at JINS (Okazaki and coworkers). Two-fluid model codes were developed at Hitachi Ltd (Minato and coworkers) to analyze two-phase flow in a BWR recirculation pump, and also at Toshiba Co. (Terasaka and coworkers) to study two-phase flow with obstacles in a BWR bundle such as a spacer. On the other hand, drift–flux models for large-diameter tubes and models for stratification and countercurrent flow using the CANAC-II code were proposed at MHI (Tsuge and coworkers) to predict the thermal-hydraulics in a PWR small-break LOCA.

Recently, some compact two-phase flow analyzers for personal computers have been developed in Japan. The MINI-TRAC code (developed by JAERI; Murao & Akimoto and coworkers) and the SIMA codes (by Hitachi; Minato and coworkers) were supplied to researchers in universities and colleges to compare them with various experimental data, including transient and steady-state two-phase flows with and without phase change for the verification and development of the constitutive equations. This type of activity is very beneficial and welcomed by both university or college researchers and code developers. There is great anticipation of fruitful results from this.

16. CONCLUSIONS

This survey of recent Japanese work on two-phase flow, which is limited to gas–liquid systems and was prepared on the basis of a considerable number of responses to a request for information, has revealed that our explorations into two-phase flow have been growing rapidly over recent years in many areas. Very impressive progress has been made in both the breadth and depth of fundamental studies and industrial applications as well, although it is clearly impossible to do justice to all of the work proceeding in Japan within the scope of this review. As has been often said, Japanese work is, on the whole, characterized as rather fundamental, this survey clearly confirmed this trend both in universities and in industry. However, this inclination toward basic studies will hopefully bring further developments and progress in two-phase flow science and technology in Japan, and, hence, lead to more advanced technologies.

The present authors hope that this review contains information useful to researchers working on two-phase flows throughout the world, and that it will stimulate the worldwide multiphase flow community to promote further development in international cooperation on multiphase flow studies.

Acknowledgements—The present authors wish to express their cordial gratitude to many persons and correspondents for their kind response to the request for information about studies currently underway on two-phase flows. Without their response, this review could not have been prepared. Thanks are also due to Drs K. Mishima, I. Kataoka and N. Takenaka for their help in the preparation of this review through the *Japan–U.S. Seminar on Two-phase Flow Dynamics*, Ohtsu, Japan, 1988.

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NARIAI, H. 1987 Review of the developments in numerical analysis of nuclear reactor thermal-hydraulics (II). Report of the Special Committee on Numerical Computation in Thermal-hydraulics, The Atomic Energy Society of Japan.

NARIAI, H. 1988 Review of the developments in numerical analysis of nuclear reactor thermal-hydraulics (III). Report of the Special Committee on Numerical Computation in Thermal-hydraulics, The Atomic Energy Society of Japan.

APPENDIX

List of Work Being Carried Out at Various Research Centers in Japan

Aichi Institute of Technology
Department of Mechanical Engineering, 1247 Yachigusa, Yakusa-cho, Toyota 470-03.
Correspondent: PROFESSOR O. TAJIMA.

AI-1: Flow and Heat Transfer of a Gas and Liquid Two-phase Flow in Helical Coils.
(Illustrative paper)
1. WATANBE, O. *et al.* 1986 Flow and heat transfer of a gas and liquid two-phase flow in helical coils. *Trans. JSME* **52,** 1857–1864 (in Japanese).

Central Research Institute of the Electric Power Industry
Power Plant Department, Mechanical Engineering Section, 2-11-1 Iwato-kita, Komae, Tokyo.
Correspondent: DR K. KAWAMURA.

CR-1: Heat Transfer Tube Vibration Induced by Two-phase Flow.
(Illustrative papers)
1. KAWAMURA, K. & YASUO, A. 1986 Boiling two-phase flow induced heat transfer tube vibration. CRIEPI Report 285090 (in Japanese).
2. KAWAMURA, K. & YASUO, A. 1988 Heat transfer tube vibration induced by two-phase jet flow from the tube support plate. CRIEPI Report (in Japanese).

Ehime University
Department of Industrial and Mechanical Engineering, 3 Bunkyo-cho, Matsuyama 790.
Correspondent: PROFESSOR K. FUTAGAMI.

EH-1: Inception of Boiling due to Depressurization.
(Illustrative paper)
1. MIZUKAMI, K., MASAOKA, R. & FUTAGAMI, K. 1986 Inception of boiling from a stainless steel surface due to depressurization. In *Proc. 23rd natn. Heat Transfer Symp. Japan*, pp. 181–183 (in Japanese).

Fukuoka Univeristy
Faculty of Engineering, 8-19-1 Nanakuma, Jounanku, Fukuoka 814-01.
Correspondent: PROFESSOR A. KARIYASAKI.

FU-1: The Behavior of the Bubble Plume Induced by a Nozzle into Still Water.
(Illustrative papers)
1. KARIYASAKI, A. & FUKANO, T. 1987 Shifted-cross-beam method (SCB method) for measurement of the local values of the characteristic parameters in a dispersed two-phase flow. In *Proc. 3rd Int. Symp. on Laser Anemometry*, pp. 161–165.

2. KARIYASAKI, A. & FUKANO, T. 1987 Shifted-cross-beam method (SCB method) for measurement of the local values of the characteristic parameters in a dispersed two-phase flow. *Jap. J. Multiphase Flow* **1,** 175–186.
3. KARIYASAKI, A. & FUKANO, T. 1988 Dispersion of bubbles in the bubble plume induced by a nozzle into still water. In *Proc. 6th Symp. on Multiphase Flow*, Tokyo, Japan, pp. 37–40 (in Japanese).
4. KARIYASAKI, A. 1986 Measurement of liquid velocity in bubbly flow. In *Proc. 39th Conf. JSME*, Kyushu District, Japan, pp. 49–51 (in Japanese).

FU-2: Behavior of a Single Bubble in a Liquid Flow with a Linear Velocity Profile.
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KBO-1: Frequency Response of Boiling Two-phase Flow.

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KB-1: Liquid–Liquid Direct-contact Heat Exchange.

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KG-3: Measurements of Phase Distribution and Flow Regime Transition in Vertical Two-phase Flow.

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Department of Mechanical Engineering, 2-39-1 Kurokami, Kumamoto 860.
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Correspondent: PROFESSOR S. YOSHIDA.

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Correspondent: PROFESSOR T. IKOHAGI.

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(Illustrative papers)

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Correspondent: PROFESSOR K. KAMIYAMA.

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(Illustrative papers)

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Correspondent: PROFESSOR R. OBA.

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1. OBA, R. *et al.* 1985 Cavitation observation in a jet-flow gate-valve. *Bull. JSME* **28**, 1036–11043.
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2. IKEDA, R. *et al.* 1987 Cavitation characteristics of right-angled branch ducts with rounded corners. *Trans. JSME* **53,** 2703–2708 (in Japanese).
3. ITO, Y. *et al.* 1987 Stereo-observation of cavitation bubbles on a Clark-Y11.7% hydrofoil. Presented at the *47th JSME Mtg*, Fukuoka, Japan (in Japanese).
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(Illustrative papers)
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1. OBA, R. *et al.* 1986 Stochastic behavior of desinent cavitation. *Trans. ASME Jl Fluids Engng* **108,** 438–443.
2. OBA, R. *et al.* 1982 Stochastic behavior of acoustic pressure pulses in the near-subcavitating range. *JSME Int. J.* **30,** 581–586 (in Japanese).
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THO-7: Mechanism of a High Speed Micro-jet.

Technical College of the University of Tokushima
Department of Mechanical Engineering, 2-1 Minamijosanjima, Tokushima 770.
Correspondent: PROFESSOR A. OUSAKA.

TC-1: Air–Water Annular Two-phase Flow in Horizontal and Near Horizontal Tubes.
(Illustrative papers)
1. SEKOGUCHI, K. *et al.* 1982 Air–water annular two-phase flow in a horizontal tube: circumferential distribution of film thickness. *Bull. JSME* **25,** 1559–1566.

2. FUKANO, T. *et al.* 1983 Air–water annular two-phase flow in a horizontal tube: circumferential variation of film thickness parameters. *Bull. JSME* **26**, 1387–1395.
3. FUKANO, T. *et al.* 1985 Breakdown of a liquid film flowing concurrently with gas in a horizontal line. *PhysicoChem. Hydrodynam.* **6**, 23–47.
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TC-2: Prediction of Film Thickness Distribution in Horizontal Air–Water Annular Flow.

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Department of Mechanical Engineering, 7-3-1 Hongo, Bunkyo-ku, Tokyo 113.
Correspondent: PROFESSOR M. HIRATA.

TK-1: Critical Heat Flux of Forced Convective Boiling in Narrow Channels.

Department of Mechanical Engineering, 7-3-1 Hongo, Bunkyo-ku, Tokyo 113.
Correspondent: PROFESSOR M. SHOJI.

TKS-1: Studies of Minimum Heat Flux and Film/Transition Boiling.
(Illustrative papers)
1. SHOJI, M. *et al.* 1987 Effects of size and end conditions of a heated surface upon minimum film boiling. *JSME Int. J.* **30**, 1587–1594.
2. SHOJI, M. & NAGANO, H. 1986 An experimental study of minimum heat flux of saturated pool boiling on a horizontal circular disk. *Trans. JSME Ser.B* **52**, 2431–2436 (in Japanese).
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(Illustrative papers)
1. SHOJI, M. *et al.* 1986 Small-scale vapor explosions of molten tin dropped into thermally stratified water (on the effect of coolant subcooling). *Trans. JSME Ser.B.* **52**, 2633–2639 (in Japanese).
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Department of Mechanical Engineering, 7-3-1 Hongo, Bunkyo-ku, Tokyo 113.
Correspondent: PROFESSOR T. UEDA. (Present address: Department of Mechanical Engineering, Kogakuin University, 1-24-2 Nishi-shinjuku, Shinjuku-ku, Tokyo 160.)

TKH-1: Studies on Heat Transfer and Flow Characteristics in Subcooled Flow Boiling.
(Illustrative papers)
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Department of Mechanical Engineering, 7-3-1 Hongo, Bunkyo-ku, Tokyo 113.
Correspondent: PROFESSOR Y. MATSUMOTO.

TKM-1: Nonlinear Behavior of Acoustic Cavitation.
(Illustrative paper)
1. MATSUMOTO, Y. 1987 Internal phenomena of acoustic cavitation bubbles. In *Proc. 11th Int. Symp. on Nonlinear Acoustics*, pp. 91–95.

TKM-2: Study on the Internal Phenomena of a Bubble.
(Illustrative papers)
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2. MATSUMOTO, Y. 1986 Contribution of homogeneous condensation inside cavitation nuclei to cavitation inception. *J. Fluids Engng* **108,** 433–437.
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Department of Mechanical Engineering, 7-3-1 Hongo, Bunkyo-ku, Tokyo 113.
Correspondent: PROFESSOR H. OHASHI.

TKO-1: An Experimental Study of the Performance of Cascade of Blades in Two-phase Flow.
(Illustrative paper)
1. OHASHI, H. *et al.* 1988 Characteristics of a circulating two phase flow tunnel. In *Proc. 6th Symp. on Multiphase Flow*, Tokyo, Japan (in Japanese).

TKO-2: Numerical Study of Two-phase Flow.
(Illustrative papers)
1. MATSUMOTO, Y. *et al.* 1987 Performance of cascade of blades in two phase flow. In *Proc. Cavitation and Multiphase Flow Forum (ASME),* Cincinnati, Ohio, U.S.A.
2. MATSUMOTO, Y. *et al.* 1987 Propagation of pressure waves in bubbly flow with size and number density distributions. In *Proc. 16th Int. Symp. on Shock Waves and Shock Tubes*, Aachen, F.R.G.
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TKO-3: Transient Phenomena in a Draft Tube during a Load Rejection of a Water Turbine.
(Illustrative paper)
1. MATSUMOTO, Y. *et al.* 1986 Transient characteristics of swirl flow in a draft tube. In *Proc. IAHR 13th Symp.*, p. 26.

Tokyo Institute of Technology
Department of Mechanical Engineering, 2-12-1 Ohokayama, Meguro-ku, Tokyo 152.
Correspondent: PROFESSOR K. HIJIKATA.

TIH-1: Convective Boiling Heat Transfer of a Mixture of Two Immiscible Liquids.
(Illustrative paper)
1. HIJIKATA, K., ITO, H. & MORI, Y. 1987 Convective boiling heat transfer of a mixture of two immiscible liquids. In *Heat Transfer Science and Technology*, pp. 401–408. Hemisphere, Washington, D.C., U.S.A.

TIH-2: A Study on Flow Characteristics and Heat Transfer in Countercurrent Water and Air Flows.
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(Illustrative paper)
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TIH-4: A Study on Instability of a Falling Liquid Film in a Countercurrent Annular–Mist Flow.
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Department of Mechanical Engineering, 2-12-1 Ohokayama, Meguro-ku, Tokyo 152.
Correspondent: PROFESSOR M. NAKAGAWA.

TIN-1: Two-phase Fluid Transient of a Bubbly Cavity Train.
(Illustrative papers)
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Research Laboratory for Nuclear Reactors, 2-12-1 Ohokayama, Meguro-ku, Tokyo 152.
Correspondent: PROFESSOR Y. KOZAWA.

TIK-1: Transient Boiling Heat Transfer in a Narrow Channel (Fluctuation Phenomena in the Channel Due to Bubble Growth and Collapse).
(Illustrative paper)
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Research Laboratory for Nuclear Reactors, 2-12-1 Ohokayama, Meguro, Tokyo.
Correspondent: PROFESSOR A. INOUE.

TII-1: MHD Pressure Drop and Heat Transfer of Helium–Lithium Annular–Mist Flow.
(Illustrative papers)
1. INOUE, A. *et al.* 1986 Studies on MHD pressure drop and heat transfer of helium–lithium annular–mist flow (2nd). In *Proc. 1986 A. Mtg of the Atomic Energy Soc. of Japan*, Paper C67, p. 217 (in Japanese).
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3. INOUE, A. *et al.* 1987 Studies on MHD pressure drop and heat transfer of helium–lithium annular–mist flow in a transverse magnetic field. *JSME Int. J.* **30,** 1768.

TII-2: Studies on Thermohydraulic Behaviors During a Power Transient and Vapor Explosion Induced by Pressure Shock.
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1. TOBITA, Y. *et al.* 1985 A study on the vapor explosion of a single drop induced by pressure shock. In *Proc. 22nd natn. Heat Transfer Symp. Japan*, Paper A212, p. 85 (in Japanese).
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TII-3: Study on Non-boiling and Boiling Two-phase Flow in Porous Solid Particle Media.
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TII-4: Vapor Condensation Phenomena on the Free Surface of Liquid Flow.
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1. TAKAHASHI, K. *et al.* 1985 Studies on condensation heat transfer to the free surface of turbulent film flow. In *Proc. 22nd natn. Heat Transfer Symp. Japan*, Paper C109, p. 323 (in Japanese).
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(Illustrative papers)
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TII-6: Basic Study on the Thermo-hydraulic Characteristics in a High Conversion Reactor Core.

TII-7: Basic Study on Two-phase Critical Flow and Two-phase Jets with Vaporization.
(Illustrative paper)
1. HAMADA, H. *et al.* 1985 Basic study on two-phase jet flow with vaporization of liquid droplets. In *Proc. 1985 A. Mtg of the Atomic Energy Soc. of Japan*, Paper F12, p. 288 (in Japanese).

TII-8: Studies on the Flow Characteristics of Bubbly Two-phase Flow Around an Obstacle.
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2. YOKOSAWA, M. *et al.* 1986 Studies on two-phase cross flow. Part II: transition Reynolds number and drag coefficient. *Int. J. Multiphase Flow* **12,** 169–184.
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Correspondent: DR T. KAGAWA.

TOS-1: Evaporative Two-phase Flow Heat Transfer of Non-azeotropic Mixtures.
(Illustrative papers)
1. OGAWA, N. & HASHIZUME, K. 1986 Pool boiling heat transfer of non-azeotropic mixtures. In *Proc. 23rd natn. Heat Transfer Symp. Japan*, Paper B134 (in Japanese).
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(Illustrative papers)
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TOS-3: Spray Heat Transfer Test During a BWR LOCA.
(Illustrative papers)
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TOS-4: Measurement of Void Distribution in Transient Two-phase Flow by an X-ray CT Scanner.
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1. NARABAYASHI, T. *et al.* 1984 Measurement of void distribution in transient flows. *Bull. JSME* **27,** 2204–2210.
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TOS-5: Two-phase Flow Analysis in a Channel with an Obstacle.
(Illustrative papers)

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TOS-6: Critical Power Analysis Using a Two-phase Multi-fluid Model.
(Illustrative papers)

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TOJ-1: Proving Test on the Thermal-hydraulic Design Reliability of a BWR Fuel Assembly (*Under contract with the Institute of Nuclear Safety, Japan, Nuclear Power Engineering Test Center*).
(Illustrative papers)

1. YOSHIDA, H. *et al.* 1986 Proving test on the thermal hydraulic design basis reliability of a BWR fuel assembly. In *Proc. 2nd Int. Top. Mtg on Nuclear Power Plant Thermal Hydraulics and Operation*, Tokyo, Japan.
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TOJ-2: BWR Rewetting Phenomena (*Joint research project with the Tokyo Electric Power Co., Tohoku Electric Power Co., Chubu Electric Power Co., Hokuriku Electric Power Co., Chugoku Electric Power Co., Japan Atomic Power Co., Toshiba Corp. and Hitachi Ltd*).

TOJ-3: Experimental Study on Void Behavior in a BWR Core (*Joint research project with the Tokyo Electric Power Co., Tohoku Electric Power Co., Chubu Electric Power Co., Hokuriku Electric Power Co., Chugoku Electric Power Co., Japan Atomic Power Co., Toshiba Corp. and Hitachi Ltd*).
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2. KASAI, S. *et al.* 1986 Heater rod temperature change at boiling transition under flow oscillation. In *Proc. 2nd Int. Top. Mtg on Nuclear Power Plant Thermal Hydraulics and Operations*, Tokyo, Japan, pp. 1.27–1.33.

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(Illustrative paper)

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