Int. J. Multiphase Flow Vol. 6, pp. 131–37
Pergamon/Elsevier, 1980. Printed in Great Britain

SOME OBSERVATIONS ON THE STATUS OF TWO-PHASE CRITICAL FLOW MODELS

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INTRODUCTION

Bouré (1978), Weisman & Tentner (1978), Lahey & Moody (1977), and Brittain (1977) have presented recent reviews on critical two-phase flow. The paper by Henry (1979), while concentrating on critical two-phase flow in nozzles, presents an excellent organization of the literature distinguishing studies involving subcooled and saturated stagnation conditions and flow geometries. Wallis (1979) develops an overview of the model developments, seeks to assess whether, for given restraints, some models are performing well enough, and comments on the degree of sophistication being used in model developments.

This paper offers a few brief observations on some selected topics dealing with the acceptability and the deficiencies in the modelling of critical two-phase flow. To prepare the reader for the particular objectives of this paper, the following points are made:

(1) There is no single, best estimate model for predicting critical two-phase flow mass rates. Even after more than three decades of research, critical two-phase flow is still an active area for research; however, there are models, as Henry (1979) notes, which appear to be adequate for carefully specified conditions.

(2) A primary need for calculating critical two-phase flow is linked with the safety and the performance of emergency core cooling systems for nuclear power reactors. As will be explained in the paper, this linkage has created some artificialities and thus some confusion in what should be the appropriate research.

(3) The special attention being given to critical two-phase flow grows out of the need for realistic predictions of the transient mass discharge rates from postulated breaks in the primary system of nuclear power reactors. Safety margins now embedded in the licensing evaluation models are believed to be adequate, but quantification of the safety margins remains to be ascertained. Even under carefully controlled experimental conditions with simulated breaks and depressurization of systems scaled to nuclear power systems, the data reduction for interpreting transient critical two-phase flows and the comparisons among models have had some problems.

(4) The paper by Wallace (1979) present judgments on which models should receive preference and which appear to be ill-advised. Though such heuristics may have merit, there is another way of addressing this problem and that is to be develop a more thorough process for model selection through comparisons with data. First, the data base for comparing model predictions needs to be reevaluated for determining the experimental errors. The selection process should present the inherent limitations used in the model, conditions for applicability, model errors, and uncertainties in the predictions. Conditions need to be agreed upon for specifying when a model is acceptable. In an example to be presented on code verification or code assurance, this process of model selection could not be completed. Other examples taken from the literature will illustrate that though comparisons of model predictions with experimental data provide an interim guide, much more work is needed to resolve acceptability of models.

(5) This paper, together with papers by Henry (1979) and Wallace (1979), serves to

formulate recommendations for continuing studies on two-phase critical flow models. The present national and international programs on critical two-phase flow do provide an expanded experimental base for model selection, and should be sufficient.

THE POSTULATED ACCIDENT

Beginning more than a decade ago, the licensing branch of the Atomic Energy Commission started the development of regulations, criteria and guides which would provide sufficient assurance that commercial nuclear power reactors could cope with a loss-of-coolant accident. Under the present Nuclear Regulatory Commission, this emphasis on safety has produced overall conservative evaluations for judging acceptability of systems which are to provide the emergency core cooling. The design basis accidents include a spectrum of postulated break sizes in the primary piping coupled with other added restraints. These postulated accidents are to be imposed abruptly by assuming that the break size is created instantaneously or within milliseconds. Models need to cope with the resulting imposed boundary conditions, and special efforts are required to simulate the appropriate breaks in the experimental systems. The postulated pipe break includes a double-ended guillotine break and an equivalent-sized longitudinal break. The use of a spectrum of pipe break sizes, along with some inoperative components and systems in the overall systems for emergency core cooling, serves to bracket the consequences of possible loss-of-coolant accidents. Obviously, an unlimited number of scenarios could be used to describe time-varying break sizes. In the past, some scoping studies indicated that no unusual conditions resulted and that the spectrum of break sizes for the design basis accidents was adequate. This aspect should be reviewed.

The critical flow at the break sets the discharge rate and strongly affects the resulting depressurization of the system. Changing fluid conditions at the break produce changes in the break flow. The imposition of a sudden, large pipe break creates difficulties in the modelling as well as in the experimental investigations. The licensing requirements have been rigid, based upon limited studies that indicate that pipe crack growth rates can be very rapid. Studies which better characterize crack growth rates and integrity of piping systems may eventually lead to some relaxation in postulated breaks used for design basis accidents. For example, it remains to be seen whether improved knowledge on the early response of hot water to sudden releases from high pressure has a significant bearing on crack growth (Lienhard *et al.* 1978, and Borkar *et al.* 1977). Further, can reliable systems be designed to assure that the very large breaks are not possible? The dilemma created is that the goal sought is the realistic or best estimate evaluation of the mass discharge rates through the break opening, but the break opening, itself, remains as an arbitrary specification.

MODELS

The papers by Henry (1979) and Wallace (1979) describe the features of critical two-phase models. In the Homogeneous Equilibrium Model (HEM), the vapor and liquid phases are uniformily distributed, each phase moves with the same velocity, and the phases are in thermal equilibrium. Furthermore, the flow process is taken to be isentropic. Nearly all models are one-dimensional in that no radial distribution effects are included. With the Moody and the Fauske models, specific relationships are introduced to provide for the slip ratio which is the vapor velocity divided by the liquid velocity. The phases need not be in thermal equilibrium and methods have been obtained in an empirical manner to account for these effects in other models. The comparisons of measured to predicted critical mass flows can be used to introduce a correction factor in terms of a multiplier for the model. Fully mechanistic approaches which attempt to follow the nucleation and growth of vapor bubbles are still dependent upon using empirical parameters which are fitted to the critical flow data and have not been independently determined or verified by basic tests. Finally, systems of equations which provide for the mass, momentum, and energy balances of each phase, together with the necessary constitutive relationships describing phase interactions and transfer between phases, have been used in advanced codes for analyzing loss-of-coolant accidents for nuclear power reactors. These systems, too, can calculate two-phase critical flow at the breaks.

SIZE EFFECT

An important question concerns the relationship of experimental observations on small diameter test sections to the large diameter nozzles and pipes used in commercial reactors. The general impression is that the critical mass velocity decreases with increases in flow area, at least for flows at low qualities.

Allemann (1979) continues to emphasize his early findings which indicated that the observed decrease in critical mass velocity with increase in flow area for break orifices was related with an increase in quality, and that the correlations of critical mass velocity vs (break area/vessel area) and critical mass velocity vs quality were similar. He suggested "...that the size effect observed for mass velocities during low quality blowdown experiments is the result of the number of bubbles which can be triggered to grow and be present as quality in the entrance region ahead of the critical flow plane. The triggering of the bubbles is directly associated with the cavitation number (the propensity not to cavitate) due to a velocity head or turbulence in the duct. Other factors which affect the creation of bubbles, such as nucleation sites, radiation, dissolved gases are involved but are influenced in the same way by the cavitation number effects." Unfortunately, his observations have not been adequately addressed in the bubble growth models. Several recent studies are noted below.

Ardron (1977) has used a one-dimensional two-fluid model for estimating critical two-phase flow of critically saturated or subcooled water discharging from a pipe. The effects of nonequilibrium between the liquid and vapor and different phase velocities are included. The evaporation rate is linked to the rate of appearance of bubbles and their rate of growth. Comparison of predictions with experimental mass fluxes for data obtained by Zaloudek *et al.* and for Fauske (limited to $L/D = 40$) gave encouraging results, using a nucleation site density of 1000/kg and an incipient boiling superheat of 3.0°. The velocity ratio of gas to liquid at the exit was never evaluated to exceed 1.25. Predictions were extended to large diameter pipes (up to 500 mm) and for L/D ratios up to 10. Under these conditions, the critical mass flux predictions for a given length of pipe was almost independent of L/D, exhibiting only a very weak maximum. Thus the flux appears to be only weakly dependent on pipe diameter.

Mather (1978) reported on the HUBBLE-BUBBLE 1 computer program which is based upon conduction controlled bubble growth theory. Apparently, this model overpredicts the discharge when compared to the tests conducted at AWRE Foulness, and thus needs to be improved. The AWRE Foulness tests represent an extension of the Edwards tests, the first of which has been widely used as a standard problem. Tests have been completed on pipe diameters up to 206 mm, but the reports have not been issued. Two-dimensional effects were noted for the large pipe.

The experimental studies of Kevorkov *et al.* (1977) for the discharge of saturated water from straight tubes with a sharp inlet edge indicate that the absolute value of the length, rather than the relative length of L/D, is the appropriate correlating parameter. Thus critical mass velocities with D varying from 14 to 37.8 mm, for saturated water at given pressures, were represented by single curves. In these tests, L/D ranged from 4 to 48.

It is recognized that where comparisons have been made of experimental data with the homogeneous equilibrium model (HEM) that the observed values could be less than those of the HEM. For example, at higher pressures and depending upon the length of the channel, the neglect of the frictional losses in the channel can produce such effects. Such is the case for the data given by Kevorkov *et al.* (1977). Further, as noted by Ardron (1977), the fit of the Fauske data for $L = 254$ mm and $D = 6.35$ mm by the HEM which was better than that from his model was considered to be fortuitous. The Ardron model does not account for sharp-edged entrances which could result in an annular flow pattern where the inner low void two-phase jet needs to travel several diameters before filling the pipe. Ardron suggested that by using a long pipe with a sharp-edged entrance the separation effects could be ignored and thus the model could be applied.

For a joint meeting on exchange of nuclear regulatory information between the U.S. and Japan, the Nuclear Safety Bureau, Science and Technology Agency, Japan (1978) concluded that the homogeneous equilibrium model, using correctly calculated inlet stagnation conditions, is sufficiently conservative for bounding boiling water reactor loss-of-coolant accidents. Included is the recognition that most critical flow experiments have utilized pipe diameters of less than a few inches, whereas pipe sizes for reactors will be more than 20 in. As noted previously, the mass flow rate would be expected to decrease as the area increases.

The staff of the U.S. Nuclear Regulatory Commission (NRC) is not prepared to conclude that the HEM for evaluating BWR LOCA is either conservative or a best estimate. To be sure, the NRC staff recognizes that critical flow behavior in large pipes may differ from the observations made with the small scale experiments. The NRC is participating in supporting the critical flow tests at the Marviken center in Sweden. Nozzle diameter up to 500 mm will be tested. Though the facility will not be able to match a complete range of flow conditions postulated for the design basis accidents of nuclear power reactors, the tests should be helpful. In the course of evaluating the HEM for predicting break flow rates for boiling water reactor containment analyses, Ross (1978) noted preliminary comparisons of HEM with early test data from Marviken which indicated that the flow rates were underpredicted by as much as 40%.

SEMISCALE TRANSIENT DATA

The following observations indicate some of the difficulties in obtaining and in interpreting the transient critical mass discharges from systems scaling the primary systems of nuclear power reactors. For example, SEMISCALE is scaled to be about 1/30 of LOFT, which, in turn, is scaled to be about 1/60 of a pressurized water nuclear power reactor. LOFT is the only integral test facility in the world which employs a nuclear reactor core. SEMISCALE uses electrically heated rods to simulate the reactor core. Foreign integral test facilities include the Japanese ROSA facility and the new Japanese cylindrical core test facility, the EURATOM LOBI test facility (which is about three times the size of SEMISCALE), and the Federal Republic of Germany (FRG) PKL facility. An integral test facility for the boiling water reactor is U.S. Two-Loop-Test-Apparatus, TLTA-2. Integral facilities simulate the entire primary system of a nuclear power reactor. Separate effects experiments simulate only portions of the system. A variety of blowdown test results are available from the integral tests and from some special separate effects tests. Nozzles are now used in these blowdown tests and are designed based upon the testing done by Henry using a converging-diverging nozzle. Some recent results are noted for SEMISCALE.

A detailed examination of a SEMISCALE experiment to evaluate the accuracy of the homogeneous equilibrium model (HEM), Moody, Henry-Fauske, Modified Burnell and Burnell critical flow models did not resolve in a completely satisfactory manner the discrepancies found between measurements of the critical flow and critical pressure with the predictions from the models. Hall (1977) distinguished three regimes for the transient data. The experimental data were reduced to yield stagnation conditions to be used for the input for the model calculations. In the first regime, the stagnation conditions correspond to a subcooled fluid. Second, a transition regime was identified to represent stagnation qualities from 0 to 2%. Varying flows were found in this regime, making the "...the modelling of the critical flow phenomena impossible with any of the five critical flow models..." The third regime is represented by stagnation qualities greater than 2%. Flow coefficients for the models in regimes 1 and 3 were obtained by comparing the measured and predicted discharges. Hall concluded that "The flow coefficients identified in the course of the present study are peculiar to the geometry of the nozzle used to produce the experimental data. The determination of flow coefficients appropriate for a given geometry in which critical flow will occur and the limits of applicability of specific models will require studies to be performed using critical flow data generated using a variety of flow geometries."

Hanson (1977) tested a Henry nozzle and an elongated nozzle and found that the critical discharge was smaller for the elongated nozzle. Hall (1977) suggested that the converging-diverging nozzle used in Semiscale, for which the Henry–Fauske model should be appropriate, may have an effective throat area smaller than the geometric area. Henry (1979) notes additionally for the Semiscale elongated nozzle that the pressure measurements may not be representative of the flow due to sensitivity of the pressure taps to the sharp corner immediately upstream.

Travis *et al.* (1978) suggest that they can explain the variations of the break flow multipliers needed for the SEMISCALE nozzle. A two-dimensional formulation of a vaporization model is being proposed. Limited, preliminary, and incomplete studies indicate that the flow multipliers depend upon the geometry of the nozzle—the ratio of the length to throat diameter. The degree of nonequilibrium between the phases does not affect the results significantly.

CODE COMPARISON AND ASSESSMENT

As noted previously, the critical mass discharge from the break initiates the sequence of events to be followed in the postulated loss-of-coolant accident (LOCA). Codes for evaluating LOCA include critical flow models. For example, Hughes & Fujita (1978) carried out a limited investigation of the critical flow models in RETRAN and compared their results with the Fauske steady-state data on critical flow in unheated, straight round tubes. RETRAN includes the Henry and Moody critical flow models and system of equations which also can be used to predict critical flow. The three models overpredicted the pressure gradients by as much as 55% when using identical flow. Contributing factors identified in the comparisons include the use of the two-phase frictional pressure drop multiplier and the manner in which the Moody and Henry choking models were used to obtain the derivatives of the mass flow rate with respect to pressure. For higher qualities (>27%), the sonic choking obtained using the system of equations appeared to give results similar to those obtained using the Moody model. A nodalization study was made for one comparison of the Moody model with one of Fauske's tests. Significant increases in the pressure gradient near the outlet end were obtained as the number of nodes was increased. This interim study serves to highlight areas that need more attention.

The independent code assessment of the RELAP 4/MOD 6 Thermal Hydraulic Transient Code has been undertaken by the Code Assessment Branch, EG&G Idaho, Inc. (1978). Among the many tasks undertaken is one which involves critical flow. In the development of the code, and for applications involving the standard Semiscale MOD-1 break simulator, the recommendation was to use the Modified Burnell critical flow model with a transition to HEM at a quality of 0.02. Multipliers were assigned to the models. An inverted analysis to define the multipliers yielded results which started at 0.7 and ended at 1.26. This unexpected large variation (with blowdown time) raised questions on the credibility of the critical flow model. The independent code assessment review involved a number of studies, but the results were inconclusive regarding recommendations for critical-flow multipliers. More experimental work was suggested. Though preference was given to the Modified Burnell model for the subcooled and quality-transition regions, with a transition to HEM at a fluid quality of 0.02, further review was recommended. Data were not available in a form for determining experimental errors and a quantification of the code uncertainties could not be made. Further, the applicability of the code to commercial nuclear power plants was not included in the assessment.

To date, no model has achieved the status of being independently assessed and approved. During model development, limited comparisons are made and improvements are achieved. Publication of such results in professional journals and reports serves a useful purpose of bringing the technical community up to date on approaches and on results. Obviously, not all models will reach the acceptance stage. Not even the details of what may be required for an independent code assessment have been agreed upon by the NRC and by the participants from the nuclear industry. There are least four features. First, procedural requirements need to be stated to assure the independence of the assessors from the code developers, and use of some experimental data not already used in the code development. Second, the data base with quantification of the experimental errors needs to be available so that uncertainties in the code, including any code biases, can be ascertained. Selection of key performance parameters for the LOCA analyses need to be made for these comparisons. Third, the applicability of the model to a full scale nuclear system needs to be achieved through the interpretation and testing of the scaled parameters. Finally, what constitutes an acceptable model? For example, the uncertainties in the critical two-phase flow model may yield a range in a selected key performance parameter, or in several such parameters. Decisions need to be made on "how good is good enough". More experience in this matter is needed, and the critical two-phase flow is only a small part of the overall needs.

RECOMMENDATIONS

Although there are some models that can be used to predict critical two-phase flow discharges in a satisfactory manner for defined conditions, several issues remain to be resolved. These issues are not with any need for evolving new critical flow tests, but are confined to improving methods for establishing comparisons of model predictions with data, and relating the conditions for using the models in LOCA analyses.

(1) The AWRE Foulness tests on pipe diameters up to 206 mm, and the Marviken nozzle tests on diameters up to 500 mm need to be reviewed to determine whether there are scaling effects and/or special dimensional effects. The applicability of these results to commercial sized plants needs to be made.

(2) The characterization of the errors in measurements for both steady state and transient critical flow for the worldwide experiments needs to be made, at least for the important sets of data. A data base is needed for evaluating models.

(3) Better definition is needed of what uncertainties are attainable and reasonable for best estimate critical flow models when applied to the postulated loss-of-coolant accidents.

(4) Complete and independent assessments need to be made of codes that are used to predict critical flows in LOCA analyses that seek to provide realistic or best estimate results.

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