EFFECTS OF STEAM-WATER VS. AIR-WATER FLUID PAIRS AND PRESSURE ON FLOODING IN LARGE-DIAMETER TUBES

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ABSTRACT

Under postulated accident conditions in a light water reactor, steam-water flooding can have important effects on the redistribution of mass and energy through the primary system. Many data exist for air-water flooding in large-diameter vertical tubes and at various pressures. Since steam-water flows are of interest for reactor applications but insufficient data are available to enable reliable analysis, the need exists to evaluate the applicability of the air-water data to steam-water applications. Factors that could produce important differences between air-water flooding and steam-water flooding include the large phase-change effect in steam-water flooding when the water is subcooled and the differences in thermodynamic properties of the fluids.

A program to experimentally compare air-water flooding data and steam-water flooding data is presented herein. First, air-water data and steam-water data obtained to develop flooding models for reactor safety codes are reviewed and the major scaling deficiency in associated facilities, namely pressure scaling, is discussed. Second, a new program to obtain data for both fluid combinations in the same experimental facility is described. The facility description and the technical approach to obtain consistent flooding data for the two fluid mixtures at up to 0.4 MPa pressure are presented.

KEYWORDS
Flooding, countercurrent flow limitation, condensation, scaling

1. INTRODUCTION

A study on flooding, or countercurrent flow limitation (CCFL), in a simple, well-characterized test section geometry, with the major variables of working fluid pairs, extent of condensation and test section pressure is presented herein. The fluid pairs used are air-water and steam-water and the flow regime is annular flow in a large-diameter vertical tube. Flooding can occur in two-phase annular flow regimes when a falling liquid film forms on the inner surface of the pipe wall and surrounds a gas-phase core in upward flow. Induced by momentum transfer from the rising gas core to the falling liquid film, onset of flooding is defined as the partial or complete flow reversal of the liquid film.

Many data exist for air-water flooding in large-diameter vertical tubes. Since steam-water flows are of interest for reactor applications but the available data are insufficient to develop reliable analysis methods, the need exists to evaluate the applicability of the air-water data to steam-water applications. The available data for the effect of pressure on flooding are also limited and the conclusions drawn by previous researchers based on these data sets are inconsistent. Yet, proper consideration of the pressure effect is essential to accurate flooding predictions.

In this paper, the available data on the effect of steam vs. air as the gas-phase working fluid and the effect of pressure on the conditions at the onset of flooding will be critically assessed. Next, a new research program to experimentally and analytically clarify these effects will be introduced. The facility description and the technical approach to obtain consistent flooding curve data for the two fluid mixtures at up to 0.4 MPa will be proposed. The originality in this paper includes, first, a comprehensive survey of available data on the effects of air-water vs. steam-water fluid pairs and of pressure on flooding occurrence in large-diameter vertical tubes with annular flow and, second, a technical approach to identify and analytically resolve any differences between air-water and steam-water flooding data over a range of pressures.

2. FLOODING IMPORTANCE TO NUCLEAR REACTOR SYSTEMS

The phenomenon of flooding can have important safety implications to light water reactor systems because, if a countercurrent flow limitation were to occur under postulated accident conditions, steam-water flooding would influence the redistribution of mass and energy throughout the primary system. At lower pressures, water injection into the reactor core will be impeded if the vaporized coolant has sufficient momentum to reverse liquid coolant direction, thereby reducing the effectiveness of safety systems. At higher pressures, flooding may affect pipe wall temperatures and the time to structural failure under hypothetical severe accident conditions.

Several scenarios in which flooding could have detrimental effects on reactor safety have been postulated. Under Design Basis Accident (DBA) conditions, one example is during the refill stage of a Pressurized Water Reactor (PWR) after a Loss of Coolant Accident (LOCA). The refill stage occurs after the initial blowdown the core coolant inventory. Emergency Core Coolant System (ECCS) water is injected via the cold legs into the downcomer, to refill the Reactor Pressure Vessel (RPV) lower plenum and to restore sustained core cooling. The ECCS water flow may be impeded by steam generated due to heat transfer from superheated metal components to the water and also by flashing of lower plenum water in the depressurized system.

Vallée et al. [2009, 2012] conducted extensive studies on flooding in the hot leg and steam generator inlet where the horizontal geometry and complex steam generator inlet geometry were carefully preserved. The hypothetical scenario of interest was a small break LOCA in a PWR, with partial failure of the high pressure ECCS and loss of forced circulation by the reactor coolant pumps. The primary system is designed to enter into a two-phase natural circulation cooling mode between the RPV and the steam generators if the RPV water level falls below the hot leg elevation. Steam flows from the RPV to the steam generators and condensate water returns from the steam generators to the RPV in horizontally stratified countercurrent flow. The condensate water is due to reflux condensation occurring in the steam generators. Vallée et al. observed that this natural circulation flow pattern is not stable under certain flow rates due to countercurrent flow limitation. Knowledge of these flooding conditions is essential because flow conditions that impede this natural circulation also limit the capability for adequate core cooling. Their experimental data, to be described in a later section, reveal the effect of phase change and pressure on flooding for the particular geometry and flow conditions.

During postulated small break LOCA scenarios, flooding can be of concern for passive PWR designs in which the Automatic Depressurization System (ADS) valves are located at the top of the pressurizer [Takeuchi, 1999]. The pressurizer in a PWR is connected to the hot leg of the main coolant system via the water-filled pressurizer surge line. The strategy is to depressurize the primary system such that water pools in the containment can drain into the RPV and maintain long-term core cooling. The piping in the surge line has sections of vertical, horizontal and slightly horizontal orientations and also includes vertical elbows. Of these piping geometries, the vertical piping was shown to be the most prone to flooding when the relief valves are cycling. Good knowledge of the pressurizer drain rate that opposes the steam flow is needed because the gravity head of the pressurizer liquid affects the pressure in the downcomer, which in turn, determines the initiation timing and the drain rate from the containment water pools.

In addition to the above DBA scenarios, flooding can influence severe accident progression. Liao and Vierow [2005] studied flooding in a hypothetical station blackout with loss of auxiliary feedwater (TMLB') severe accident. In this postulated scenario, the coolant inventory in the RPV boils off and hot steam passes from the RPV through the hot legs to the steam generators. As the steam natural circulation flow pattern develops between the RPV and the steam generator U-tubes, several components are exposed to high pressure (about 15.5 MPa) and high-temperature (1000K, or 727^{o}C) gas. Under such challenging conditions, one or more structural components may fail. The desirable result would be to have any component other than the steam generator tubes fail first. A failure by any other component than the steam generator tubes would cause the primary system to depressurize into the containment and the concern of containment bypass through the steam generator tubes would be alleviated.

In this station blackout event, the hot steam flows past the pressurizer surge line and also through it. As coolant is discharged through the cycling relief valve at the top of the pressurizer, the possibility of flooding in the surge line increases. If flooding were to occur in the surge line, the surge line wall would be exposed directly to the hot steam. This is due to a breakdown of the annular film and formation of local dryout conditions on the pipe wall inner surface. Should dryout occur for an extended period of time, the pressurizer surge line could rupture due to plastic deformation caused by high temperature creep.

These examples illustrate the potential effects of flooding on reactor safety. Knowledge of the conditions for flooding initiation is essential for establishing appropriate accident management strategies. This paper presents a study to characterize the steam vs. air effect, the phase-change effect on flooding occurrence, and any otherwise unresolved pressure effect. The study focuses on flooding in a large-diameter vertical tube with annular flow. In this simple, well-characterized geometry, some of the basic mechanisms of flooding can be better understood without complications from complex geometry-induced flow patterns or phasic interactions, while obtaining information for direct application to reactor safety analysis.

3. PREVIOUS RESEARCH

3.1. Flooding Conditions

In presenting flooding data, the flooding curve is generally plotted in terms of a dimensionless liquid superficial velocity and a dimensionless vapor superficial velocity. The classical flooding curves developed by Wallis et al. [e.g. 1961] were for air-water experiments and were correlated in terms of the following parameters.

$$j_f^* = \frac{\rho_f^{1/2} j_f}{\left[gD\left(\rho_f - \rho_g\right)\right]^{1/2}} \tag{1}$$

$$j_g^* = \frac{\rho_g^{1/2} j_g}{\left[gD\left(\rho_f - \rho_g\right)\right]^{1/2}} \quad (2)$$

where j_f^* and j_g^* are the dimensionless superficial velocities of liquid and gas respectively, ρ is density, D is tube diameter, and g is the gravitational constant. The subscripts f and g are for liquid-phase and gas-phase respectively. These dimensionless velocities are the ratio of inertial to buoyancy forces. The Wallis correlation has the functional form of:

$$j_g^{*1/2} + m j_f^{*1/2} = C \quad (3)$$

where j_f^* is the liquid superficial velocity flowing down the tube at the onset of flooding and the constants m and C depend on the geometry of the test section and the operational procedures.

Pushkina [1969] observed that, for large-diameter tubes, the tube diameter does not have an impact on the magnitude of air velocity required to breakdown the film. A more appropriate parameter for correlating air-water flooding data than the Wallis parameter is the Kutateladze-type parameter, Ku_g and Ku_f, which replaces the length scale in the Wallis parameter with the Laplace capillary length in Equations 1-3.

$$Ku_f = \frac{\rho_f^{1/2} j_f}{\left[g\sigma\left(\rho_f - \rho_g\right)\right]^{1/4}} \quad (4)$$

$$Ku_g = \frac{\rho_g^{1/2} j_g}{\left[g\sigma\left(\rho_f - \rho_g\right)\right]^{1/4}} \quad (5)$$

The tube diameter may be classified as “large” or “small” base on whether the tube diameter influences the flooding. Smaller tubes may experience liquid film bridging, which is not possible in larger tubes. Wallis and Makkenchery [1974] demonstrated the critical value of the gas Kutateladze number to be 3.2, above which flooding characteristics are independent of tube diameter.

Tien [1977] included the effect of vapor condensation on subcooled liquid by formulating an effective vapor flow rate. Since the latent heat released in vapor condensation is absorbed by the liquid as sensible heat, the effective gas Kutateladze number, Ku_{ge}, can be expressed as:

$$Ku_{ge} = Ku_g - f\frac{c_p \Delta T_{sub}}{h_{fg}}\left(\frac{\rho_f}{\rho_g}\right)^{1/2} \quad (6)$$

where Ku_g is the gas Kutateladze number at the inlet. The fraction of vapor condensed, f, is an empirical constant.

3.2. Comparison of Air-water to Steam-water Flooding Data and Effect of Phase Change

Past papers discussing flooding in reactor safety applications are reviewed below. While this current study focuses on flooding in a large-diameter vertical tube with annular flow, much of the current knowledge about flooding and the effects of fluid pairs and pressure is obtained from studies in which the test section was of a geometry specific to a particular scenario and/or operating procedures affected the results. The analysis below attempts to glean the relevant knowledge and to highlight the new contributions that the current study is pursuing.

Rothe and Crowley [1978]

Rothe and Crowley [1978] examined the refill stage in one of the earlier studies of flooding on reactor safety systems. Focusing on the PWR refill stage, Rothe and Crowley investigated flooding as it would occur in the downcomer between the RPV wall and the core barrel. The experimental data used in this study was obtained in two scaled-down vessels representing the RPV at 1:15 and 1:30 scales. Figure 1 shows the flow paths under consideration. The fluid pair was steam and water; the test pressure ranged between 0.1 and 0.44 MPa and subcoolings between 0°C and 104°C.

The flooding data were correlated in terms of the dimensionless fluid velocities, j_g^* and j_f^*, with a factor f applied to account for condensation in complete bypass tests. The f factor was determined empirically and it is intended to reflect the "efficiency of the condensation". It corresponds to the factor proposed by Tien [1977] in Equation (6). For a fixed ECC injection rate, j_f^*, a constant value for f was found to be satisfactory for flooding correlations. However, Rothe and Crowley conclude that methods to scale condensation rates must be developed and that larger scale tests are needed to verify the small-scale data.

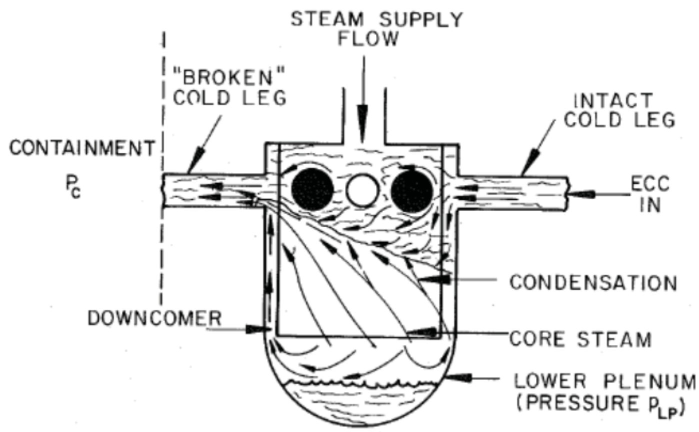

Figure 1 Geometry of Flooding During Downcomer Refill Considered by Rothe and Crowley [1978]

The "unwrapped annulus" geometry employed by Rothe and Crowley likely influences the values of coefficients in flooding correlations. Specifically, steam and water injection approaches in these experiments were different than in the vertical tube geometry and annular flow was likely not achieved in the tests evaluated by these earlier authors. Therefore, the gas-liquid interactions differ from those in a large-diameter tube with annular flow. Their conclusions are restricted to the geometry of ECCS injection into a downcomer. The data to be obtained in this current study will utilize a fundamentally simpler geometry and are expected to reveal more of the basic physics of the flooding phenomena than could be obtained from the ECCS-application studies.

Wallis [1980]

Wallis [1980] performed air-water and steam-water tests with test sections of the schematic shown in Figure 2. Based on the calculated Ku_g values for the tests, they are classified as small diameter. Water temperatures were 9°C, 15°C, 71°C and 96°C and all tests appear to have been conducted at atmospheric pressure. In addition to the small test section diameter, a major difference between the Wallis facility and the current vertical-tube test facility is the operational procedure of having water injection from an upper plenum. This water head imposes a different pressure boundary condition on the Wallis test section than the Rothe and Crowley facility and the current test facility have, as steam must pass through a water pool to exit the test section. The Wallis flooding conditions were influenced by the water level in the upper plenum and in the lower plenum.

Wallis et al. found that air-water flooding data and steam-water flooding data agree to within experimental uncertainty. They noted that, in their facility, condensation occurred almost exclusively in the upper or lower plenum and the hydrodynamics of the countercurrent flow limitation seemed independent of the condensation. If the liquid entering the location of CCFL were near saturation based on upper plenum condensation, it would explain the different conclusions between Wallis and Rothe.

Vallée et al. [2009, 2012]

Countercurrent flow limitation tests were performed in a scaled-down section of a PWR hot leg by Vallée et al. [2009, 2012]. The test section simulates the hot leg of a German Konvoi-type PWR on a 1:3 scale. Representing a length-wise cut-through along the vertical mid-plane of the hot leg and the steam generator inlet chamber, the test section is of rectangular cross section. As shown in Figure 3, flow travels through the horizontal channel of the hot leg, a bend connecting to an upward-inclined rectangular section and a quarter-segment of a circle representing the steam generator inlet plenum. The horizontal segment has a rectangular cross-section of 0.05 m width and 0.25 m height.

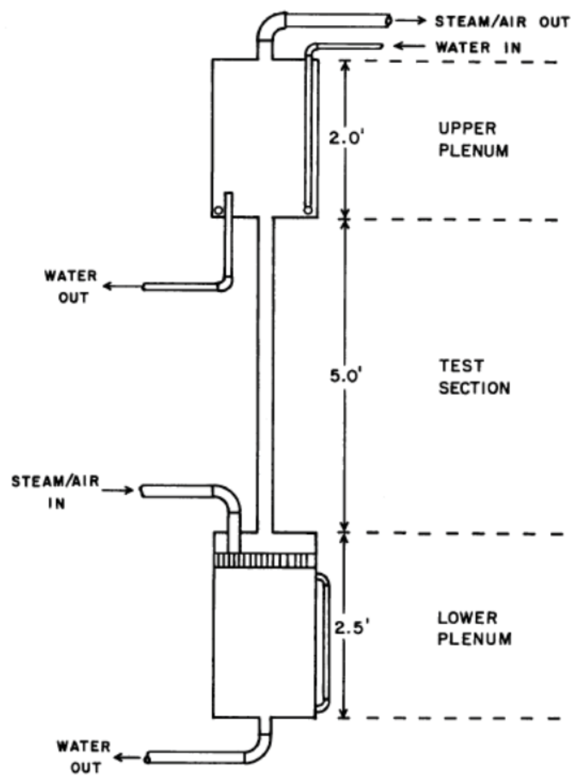

Figure 2 Schematic of the Wallis [1980] Test Facilities

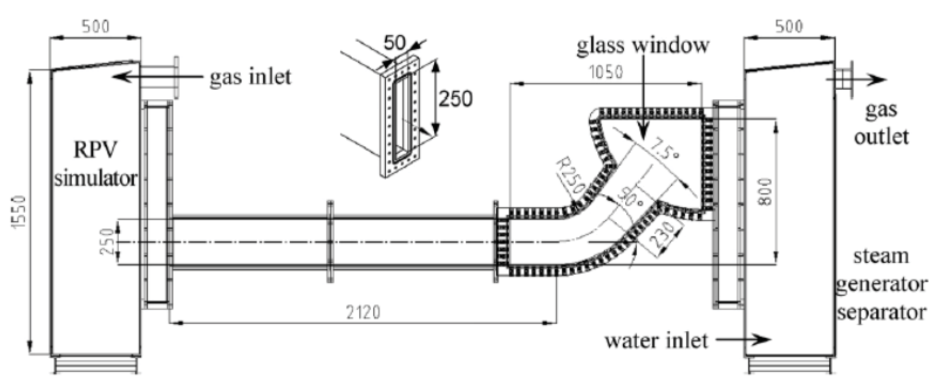

Figure 3 Schematic of the Vallée et al. [2009, 2012] Test Section (dimensions in mm)

Comparisons between air-water tests and steam-saturated water tests provide valuable confirmation that the conditions for flooding are very similar for the two working fluid pairs when there is minimal vapor condensation [Vallée et al., 2012]. Air-water tests were conducted at pressures of 0.15 and 0.3 MPa while steam-saturated water tests were performed at 1.5 MPa, 3.0 MPa and 5.0 MPa, with an unavoidable subcooling of about 2^{o}C. The Wallis-type parameter was shown to better correlate the data than the Kutateladze-type parameters for both air and steam. The later paper by these researchers corrected some of their earlier work [Vallée et al. 2009] in which discrepancies between the air-water and steam-water flooding curves were thought to have been resolved by a correction term based on fluid viscosities. The 2012 paper revealed that the inlet steam flow rate needed to be corrected for condensation that occurred between the steam flow rate measurement location and the test section inlet. After which good agreement with the base correlation was observed. In the 2009 paper, Vallee showed an apparent pressure effect in the flooding curves obtained in full-scale data from the Upper Plenum Test Facility (UPTF). This need for a pressure correction stands in opposition to their 2012 conclusions based on data from their facility.

While these conclusions about the similarity in flooding characteristics for fluid pairs provide indications of the results to be found in the current study, care must be taking in applying them to flooding in a large-diameter vertical tube with annular flow. First, the correlating parameter is different. The Wallis parameters used in the Vallée study appear appropriate for horizontal, stratified flow in rectangular channels. However the shape and amount of gas-liquid interface is different from that in vertical, circular channels with annular flow and there is a strong technical basis for using the Kutateladze-type parameter for vertical channels with annular flow (see Ritchey and Cullum results below). Second, Vallée et al. [2012] discussed the uncertainty in the proper characteristic length to use in the Wallis parameters for the hot leg application. They chose the dimensions that gave the best agreement with the UPTF data. Third, flooding characteristics are certainly strongly influenced by the various geometries associated with the hot leg and steam inlet plenum. Therefore, the results of these hot leg studies are taken as indications of trends to be seen in future testing, but the conclusions cannot be assumed for the current application without verification.

<u>Ritchey [2011]</u>

Ritchey [2011] performed flooding experiments in a 0.076 m inner diameter vertical tube with steam and water. The motivation for these experiments and the Cullum [2014] work described below derives from the concern for flooding in the PWR pressurizer surge line under postulated severe accident conditions and it also directly addresses the issues raised by Takeuchi [1999] for flooding in the vertical section of the surge line of the pressurizer during ADS activation. The inlet water temperature in the Ritchey tests was 70^{o}C (30^{o}C subcooling) and all tests were performed at atmospheric pressure. The water inlet was designed to assure annular flow in the test section and water injection procedures appear equivalent to the Rothe and Crowley type of water injection, that is, without an imposed water head in an upper plenum.

Ritchey found that, when plotting the flooding data in terms of the gas and liquid Kutateladze numbers, a correction based on the method suggested by Tien [1977] to obtain an effective gas Kutateladze number was necessary. The Tien formula shown in Equation (6) had to be modified because the steam was slightly superheated and the water was always subcooled in Ritchey's experiments. Equations (7) and (8) show Ritchey's formulation for the effective Kutateladze number and fraction of steam condensed between the test section inlet and the location of flooding.

$$Ku_{ge} = \frac{\rho_g^{1/2} j_{ge}}{\left[g\sigma\left(\rho_f - \rho_g\right)\right]^{1/4}} = Ku_g\left(1 - f\right) \tag{7}$$

where j_{ge} is the effective superficial gas velocity and the fraction of steam condensed, f, is calculated as

$$f = \frac{\dot{m}_f c_p \left(T_{exit} - T_{wall}\right)}{\dot{m}_{st}\left(h_{st} - h_f\right)} \quad (8)$$

where the numerator represents the energy gained by the water and the denominator is the total energy necessary to condense all of the steam.

The flooding data were fit with the following empirical correlation.

$$Ku_{ge}^{1/2} + 0.56 Ku_f^{1/2} = 1.6 \quad (9)$$

<u>Cullum [2014]</u>

Cullum continued Ritchey's work by modifying the test facility to enable testing over a range of water inlet subcooling between 3^{o}C and 65^{o}C. The same methods for data analysis were applied and flooding curve constants that better captured the subcooling/condensation effect on flooding were obtained as in Equation (10):

$$Ku_{ge}^{1/2} + 0.37 Ku_f^{1/2} = 1.44 \quad (10)$$

Cullum also performed an important benchmark by comparing the nearly saturated (3^{o}C subcooling) steam-water data with air-water data by Ritchey (née Williams) [Williams, 2009], as shown in Figure 4. The air-water data were obtained by Williams at room temperature. The steam-water data should match the air-water data if there is no phase change, unless temperature affects the air data through material property differences. The good match between the air-water data and the steam-water data establishes the nearly saturated steam-water data as a valid reference set against which to quantify the effect of water inlet subcooling in the rest of Cullum's data.

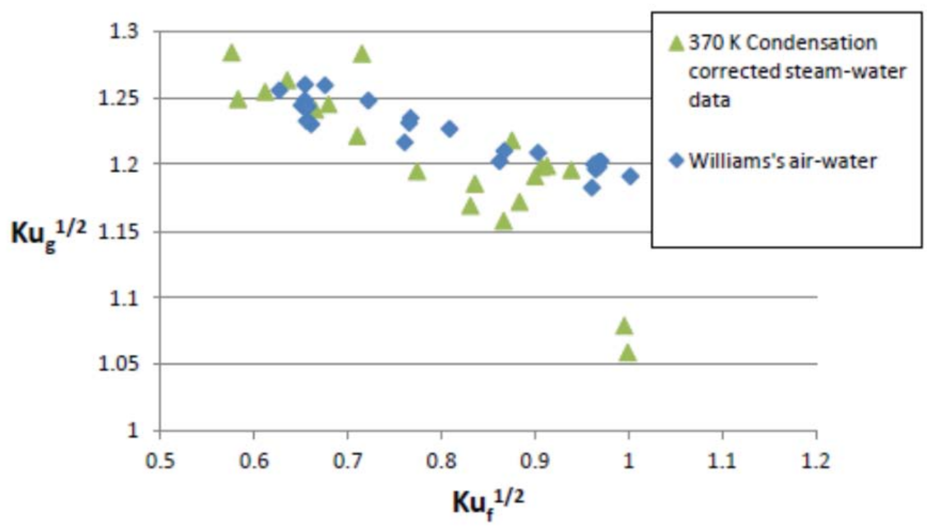

Figure 4 Comparison between Nearly Saturated Steam-Water and Air-Water Data [Cullum, 2014]

In assessing the collective data bases presented above, one realizes that each test program had particular geometries or conditions imposed upon it which prevent consolidation of the data into a single data base. Prior to the work of Ritchey and Cullum, the experiments were performed with particular reactor safety applications in mind and these data have some restrictions in their use.

The testing program described herein aims to produce quality data for air-water and steam-water comparison in a simple and well-characterized facility. The simplicity in geometry removes some of the restrictions in applicability and renders the data generic to large-diameter vertical tube geometries.

Ritchey's data analysis methods will be employed to estimate the fraction of condensed steam. A better estimate of the location of flooding within the test section will be attempted as this determines the temperature measurement used in the calculation of the fraction of condensed steam, and therefore a better estimate of the local effective gas superficial velocity. In the current work, the test section used in the Cullum experiments is being upgraded to enable tests with water at an inlet temperature up to saturation temperature at 0.4 MPa.

Key contributions from the current test program are expected to be comparisons of air-water data with steam-saturated water data from the same test section and the assessment of the subcooling (i.e. condensation) effect over a large range of water subcooling. Comparison of the data will enable benchmarking of the steam-water data to air-water data and better understanding of steam-water flooding when condensation occurs.

3.3 Effect of Pressure on Flooding Data

Pressure is expected to affect flooding for several reasons. Considering the gas-water interactions, the water-to-steam density ratio is much smaller at higher pressure. At 0.1 MPa, the ratio is about 1600:1, while at the PWR operating pressure of about 16 MPa, the ratio is about 5.5:1. At the upper pressure limit of several testing programs described herein, the ratio is 424:1 at 0.4 MPa. The momentum exchange between the phases should be of a different order of magnitude as the density ratio changes significantly. That is, relatively denser steam should be more able to reverse the water flow direction and thereby affect conditions for the onset of flooding.

Rothe and Crowley [1978]

Rothe and Crowley found that, over the range of pressures for their reported data, the flooding data plotted in terms of the Wallis parameters for saturated water tests is insensitive to pressure over the pressure range of 0.1 to 0.44 MPa. Stated another way, the pressure related terms in the Wallis formulation were sufficient to capture the pressure effect in their experiments in a complex geometry. Figure 5 shows sample results of steam and nearly-saturated water tests at between 0.1MPa and 0.44 MPa on which this conclusion is based. Rothe and Crowley added that more data needs to be taken over larger scales to validate the small-scale tests.

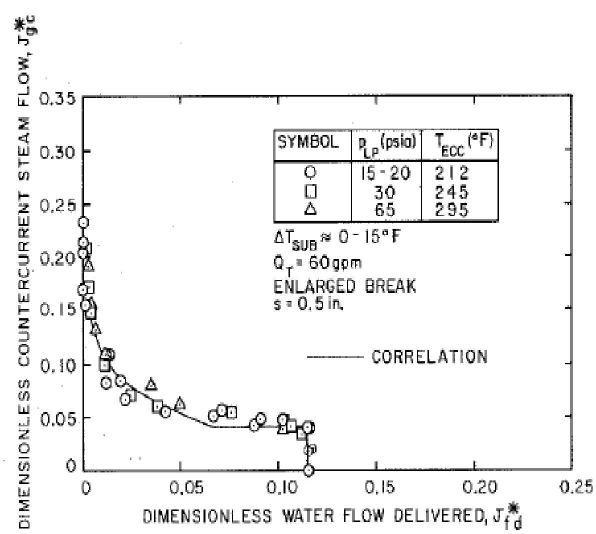

Figure 5 Rothe and Crowley [1978] Tests Showing Negligible Pressure Effect on Flooding

Vallée et al. [2009, 2012]

The Vallée et al. [2009, 2012] data provide the highest pressure flooding data found in the literature. After making corrections for steam condensation upstream of the test section entrance and liquid entrainment from the separator vessel, flooding data showed negligible effects of the pressure from 0.15 MPa to 5.0 MPa.

According to the Vallée et al. [2012] report, thermodynamic properties used in the Wallis parameter are sufficient to capture the pressure effect for the data from Upper Plenum Test Facility program. Applicability to flooding in round tubes and annular flow remains to be verified.

Ilyukhin et al. [1999]

Ilyukhin et al. [1999] performed adiabatic flooding tests in a vertical tube test section over a pressure range of 0.3 to 1.6 MPa. The hydraulic diameters of the two test sections were 0.03 m (circular) and 0.04 m (square) respectively. The working fluids were steam and water and the water was maintained at saturation temperature. Therefore, these tests could be of valuable reference in isolating the effect of pressure on steam-water flooding.

The data are correlated in terms of the Kutateladze number, which the current authors (Vierow et al.) believe is incorrect for small diameter tubes as based on Figure 6, Ku_g is below 3.2 which is the criteria for small diameter. The Wallis parameter is more appropriate for the Ilyukhin data because the tube diameter effect should be present. Figure 6 shows the flooding data in terms of "complexes U^*_i", which are not defined in the 1999 paper but are believed to be the Kutateladze parameter combined with the pressure correction and other factors proposed by Ilyukhin et al. Note that the data are plotted in reverse of convention. That is, in Figure 6, the liquid complex U^*_1 is on the y-axis and the gas complex U^*_2 is on the x-axis.

The pressure correction term is determined from Equation (4) of the Ilyukhin et al. [1999] paper to be:

$$Pressure\ correction\ term = Bo^{0.125}\left(\frac{\rho_g}{\rho_f}\right)^{0.05} \tag{11}$$

where Bo is the Bond number. It is noted that what is portrayed as a pressure correction term appears to primarily transitions the Kutateladze number to the more appropriate Wallis number, This term is based on earlier data of Ilyukhin et al. over a pressure range of 1.0 MPa to 8.0 MPa, as reported in their 1999 paper. The authors appear to have assumed that this correlation describes the data over their newer testing range of 0.3-1.6 MPa. The current authors (Vierow et al.) plotted this correction term as shown in Figure 7. This plot shows that the pressure can affect the flooding data by a factor of two. Since this pressure correction results in a Wallis term, the test section hydraulic diameter is inappropriately included as a pressure related term.

Based on these four data sets, the effect of pressure on flooding is uncertain. The expected density effect is not readily apparent in most of the data. There is consensus in the field that the pressure scaling issue has been only partially addressed because data at higher pressures is needed to validate use of the lower pressure data at elevated pressure. Testing at lower pressures has value in indicating what the pressure dependence may be at higher pressures and in providing guidance to experimental test design and analysis at higher pressures. The current test program will provide such guidance by performing tests that isolate the effect of pressure on saturated fluid properties which govern flooding phenomena, as Ilyukhin et al. [1999] did. Namely, tests will be performed with steam-saturated water over a range of pressures.

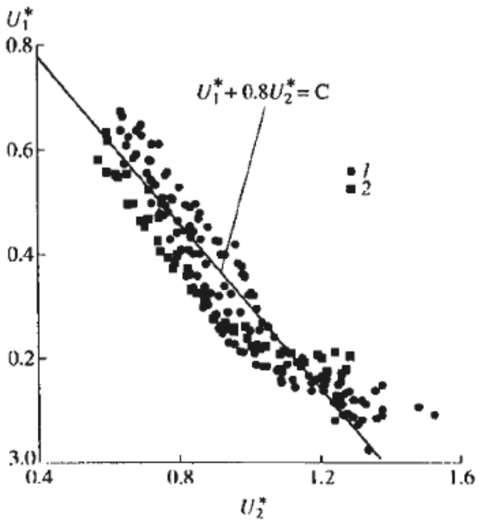

Figure 6 Ilyukhin [1999] Flooding Data at 0.3-1.6 MPa

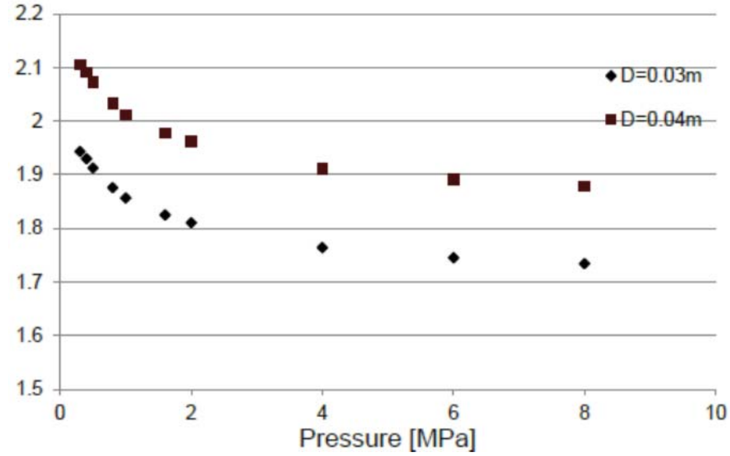

Figure 7 Pressure Correction Term Deduced from the Ilyukhin et al. [1999] Paper

4. TECHNICAL APPROACH TO RESOLVING THE EFFECTS OF FLUID PAIRS AND PRESSURE ON FLOODING

4.1 Experimental Approach

The steam-water flooding test facility at Texas A&M University was designed and built by Williams [2009]. The facility schematic with future modifications to enable steam-saturated water testing over a pressure range and air-water testing over the same pressure range is shown in Figure 8. The test section is stainless steel 304 with a height of 1.828 m and an inner diameter of 0.0762 m. At the top of the test section, the water inlet has been designed and tested to ensure formation of an annular film circumferentially around the inner surface of the test section. A 150-kW steam generator supplies steam to the test section.

The facility is equipped with state-of-the-art instrumentation to provide the user with essential data during flooding tests. The instrumentation includes thermocouples, pressure transducers, a vortex flow meter, and a magnetic flow meter. The output of each instrument is wired into a National Instruments SCXI-1102b module which is connected to a SCXI-1000 chassis. The software component of the data acquisition system is a Labview virtual interface.

For each flooding test, the water flow rate will be set to a predetermined value before the test is commenced. Then the steam flow rate will be slowly and incrementally increased until flooding occurs. Once flooding is achieved, the inlet steam flow rate will be set to remain constant for a long enough period of time to identify the conditions associated with the onset of flooding. At this point, the steam isolation valve will be closed. The test matrices for the air-water and steam-water tests are provided in Tables I and II.

Table I Range of Air-water Test Conditions

Parameter	Range
Test Section Pressure	0-0,41 MPa
Air Flow Rate	0 – 0.075 kg/s
Air Temperature	25°C
Water Flow Rate	0.00022 – 0.00076 m^3/s
Water Temperature	25°C

Table II Range of Steam-water Test Conditions

Parameter	Range
Test Section Pressure	0-0,41 MPa
Steam Generator Pressure	0,34-0.72 MPa
Steam Flow Rate	0 – 0.075 kg/s
Steam Temperature	110-155°C
Water Flow Rate	0.00022 – 0.00076 m^3/s
Water Temperature	As near to saturation as possible, 60°C, 25°C

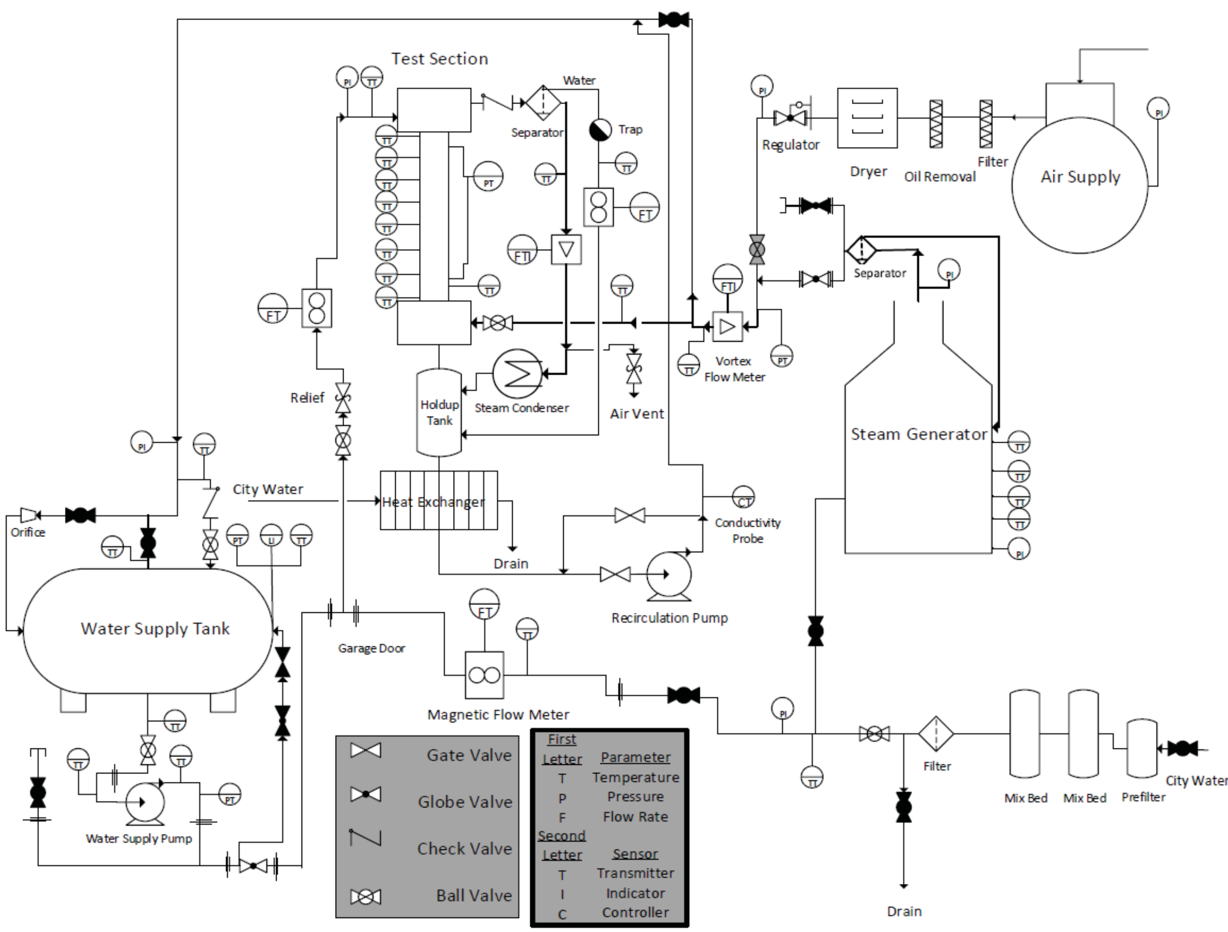

Figure 8 Piping and Instrumentation Diagram of Facility

4.2 Analytical Approach

The analysis methods for evaluating the new test data will be the same as those employed by Ritchey [2011] and Cullum [2014] (described earlier). The factor, *f*, to account for the amount of steam condensation will be tested at higher pressure when pressure is the only test variable, i.e. no variation in subcooling or change of fluid pair. As no data has been found for this factor at above atmospheric pressure, this will be a new contribution.

The next step will be to evaluate the cross-correlations of effects of fluid pairs, condensation and pressure on the flooding data. Rothe and Crowley [1978] demonstrated that the subcooling effect changes with pressure. The flooding data trends are the same at different pressures, however the magnitude of the subcooling effect decreases with pressure. Cullum [2014] showed the details of the condensation effect via a subcooling study at atmospheric pressure. Cullum's study will be extended to include the effect of fluid pairs and of pressure and will investigate the influence of the variables upon each other.

An examination of the impact of thermodynamic properties is included. Three important studies summarized below show the lack of agreement on the effects of thermodynamic properties. The current program should be able to shed some light on these aspects because the most localized and detailed data to date for large-diameter vertical tube flooding will be obtained.

Wallis [1980]

Wallis [1980] reported air-water flooding results for tests in which the water inlet temperature was between 9 and 96^{o}C at atmospheric pressure. The flooding curve in terms of the Wallis superficial velocities had the expected shape of a curve trending downwards toward the right. The slope of the curve increased with temperature. The interpretation was that the differences in the curves were due to "humidification and heating of the air by the hot water", not due to surface tension or viscosity.

Chung et al. [1980]

Chung et al. [1980] considered the physics of flooding onset and the stability of the gas-liquid interface. They postulated that an increase in liquid viscosity will allow flooding to occur at a lower air flow rate because waves on the liquid film surface are formed with smaller radii of curvature. Counteracting this destabilizing effect, viscosity also has a damping effect on interface oscillation, thereby delaying the onset of instability. The destabilizing effect of increased liquid viscosity was confirmed by experiment; however, the effect is only important when the viscosity is changed by an order of magnitude or more.

Zapke and Kroger [1996]

Zapke and Kroger [1996] refute Chung et al.'s conclusion that liquid viscosity has a small effect. They plotted Chung et al.'s data in terms of a dimensionless parameter containing liquid density, viscosity surface tension and diameter. Plotted this way, there is an impact of the parameter on the flooding curve.

5. CONCLUSIONS

A study on flooding, or countercurrent flow limitation (CCFL), in a simple, well-characterized test section geometry, with the major variables of working fluid, extent of condensation and test section pressure has been presented herein. Flooding occurs in two-phase annular flow regimes where liquid forms an annular film on the inner surface of the pipe wall, surrounding upward flow of the gas-phase core. Flooding is defined as the partial or complete flow reversal of the liquid film. The potential for flow reversal is important to nuclear reactor safety because it influences the spatial and temporal distributions of mass and energy within reactor systems and could negatively impact core cooling capabilities.
A critical assessment of past flooding studies was performed and significant discrepancies were found among previous researchers as to the effects of these parameters.

A technical approach to experimentally and analytically clarify these effects was introduced. An experimental facility to obtain flooding data in a large-diameter vertical tube with annular flow and the technical approach to obtain consistent flooding data for the two fluid mixtures at up to 0,4 MPa were described. This new flooding data will have particular value because it will be acquired in a test facility that has a well-characterized geometry and proven techniques to ensure an annular film and measure fluid properties at key locations. The fundamental physics of two-phase flooding phenomena can be studied, uncomplicated by geometry effects. Comparison of the data will enable benchmarking of the steam-water data to air-water data and therefore better understanding of steam-water flooding at different pressures and under varying degrees of vapor condensation.

REFERENCES

1. C. Vallée et al., "Counter-current Flow Limitation Experiments in a Model of the Hot Leg of a Pressurised Water Reactor – Comparison Between Low Pressure Air/Water Experiments and High Pressure Steam/Water Experiments", *Proc. 13th Intl. Top. Mtg. on Nucl. Rctr. Thermal Hydraulics (NURETH-13),* Kanazawa City, Japan, Sept. 27–Oct. 2, 2009, CD-ROM (2009).
2. C. Vallée et al., "Counter-current Flow Limitation Experiments in a Model of the Hot Leg-Comparison Between Air/water and Steam/water Experiments", *Nuclear Engineering and Design*, **245**, pp. 113-124 (2012).
3. K. Takeuchi et al., "Flooding in the Pressurizer Surge Line of AP600 Plant and Analyses of APEX Data", *Nuclear Engineering and Design*, **192**, pp. 45-58 (1999).
4. Y. Liao and K. Vierow, "MELCOR Modeling of Creep Rupture in Steam Generator Tubes", *Nuclear Technology*, **152**(3), pp. 302-313 (2005).
5. G. B. Wallis, *Flooding Velocities for Air and Water in Vertical Tubes*, AEEW-R-123, U. K. Atomic Energy Authority, Reactor Group, Atomic Energy Establishment (1961).
6. O. L. Pushkina and Y. L. Sorokin, "Breakdown of liquid film motion in vertical tubes", *Heat Transfer – Soviet Research*, **1**(5), pp. 56-64 (1969).
7. C. L. Tien, "A Simple Analytical Model for Counter-current Flow Limiting Phenomena with Vapor Condensation", *Letters in Heat and Mass Transfer,* **4**, pp. 231–238 (1977).
8. P.H. Rothe and C.J. Crowley, *Scaling Pressure and Subcooling for Countercurrent Flow*, NUREG/CR-0464, CREARE-TN-285, prepared by Creare, Inc. for U.S. Nuclear Regulatory Commission, Hanover, NH, USA (1978).
9. G.B. Wallis et al., *Countercurrent annular flow regimes for steam and subcooled water in a vertical tube*, NP-1336, prepared by Dartmouth College for Electric Power Research Institute, Hanover, NH, USA 1980.
10. M. Vijayan, S. Jayanti, and A. R. Balakrishnan, "Effect of Tube Diameter on Flooding," *Int. J. Multiphase Flow*, **27**, pp. 797-816 (2001).
11. S. N. Ritchey, M. Solom, O. Draznin, I. Choutapalli, K. Vierow, "Flooding Experiments with Steam and Water in a Large Diameter Vertical Tube", *Nuclear Technology*, **175**, pp. 529-537 (2011).
12. W. Cullum, J. Reid, K. Vierow, "Water Inlet Subcooling Effects on Flooding with Steam and Water in a Large Diameter Vertical Tube", *Nuclear Engineering and Design*, **273**, pp. 110-118 (2014).
13. S. N. Williams, *Flooding Experiments with Steam and Water in a Large Diameter Vertical Tube*, M.S. Thesis, Dept. of Nuclear Engineering, Texas A&M University, College Station, TX USA (2009).
14. Y. N. Ilyukhin et al., "The Hydrodynamic Characteristics of Process of "Flooding" under Conditions of Countercurrent Flow of Steam and Water in Vertical Tubes", *High Temperature*, **37**(3), pp. 463, (1999).
15. S. K. Chung, L. P. Liu, and C. L. Tien, "Flooding in Two-Phase Counter-Current Flows—II Experimental Investigation," *Physicochemical Hydrodynamics*, **1**, pp. 209-220 (1980).
16. A. Zapke and D. G. Kroger, "The Influence of Fluid Properties and Inlet Geometry on Flooding in Vertical and Inclined Tubes," *Int. J. Multiphase Flow*, **22**, pp. 461-472 (1996).