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INVESTIGATION OF FLASHING FLOW OF SUBCOOLED LIQUID THROUGH MICROSLITS AND CRACKS

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Abstract

The ability to estimate the leak rates from the throughwall flaws (cracks, slits, pits, frets, etc.,) is important in terms of radiological source potentially to be released into the environment as well as to the overall safe operation of nuclear power plant. In this study an experimental program and analysis methods were developed to measure and assess the chocking flow rates of initially subcooled water through micro-slits and cracks geometries. Experimental program involved chocking flow for various simulated throughwall flaw geometries for vessel pressures up to 7 MPa with various subcoolings. Measurements were performed on subcooled flashing flow through well-defined throughwall slits and cracks of thin-walled piping and vessel components with ratio of channel length to hydraulic diameter, L/D <5.5, and liquid subcoolings between 14 and 51^{0}C.

1. Introduction

When performing a leak-before-break integrity assessment for pressurized components, it is important to demonstrate that the leak is readily detectable. This assumes the ability to predict the flaw opening area and corresponding leakage rate. The ability to estimate the leak rates from the throughwall flaws (cracks, slits, pits, frets, etc.,) is required to estimate radiological source potentially to be released into the environment as well as to the overall safe operation of nuclear power plant. In this study an experimental program and analysis methods were developed to measure and assess the chocking flow rates of initially subcooled water through micro-slits and cracks geometries.

2. Choking Flow in Literature

Choking flow is a phenomenon which occurs in a wide range of industrial systems. It is especially important in a nuclear reactor; where high pressure subcooled water in the case of a CANDU and PWR is used to generate steam. In the instance of a loss of coolant accident (LOCA), choking flow determines the coolant inventory of the reactor vessel. If choking were not to occur, the reactor water inventory would be depleted rapidly. This is not the case however because, as the pressurized subcooled water nears the break, it flashes to vapor which limits the mass flow rate due to choking. Therefore, the integrity of the core during a LOCA is dependent upon this choking phenomenon.

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A LOCA from a small break or large break in the main steam line or valve is not the only place in a reactor where coolant could be escaping the primary side of the reactor. Steam generator (SG) tubes have a history of small cracks and even ruptures, which lead to a loss of coolant from the primary side to the secondary side under the previously explained leak before break approach to steam generator tube integrity. Therefore, choking flow plays an integral part not only in the engineered safeguards of a nuclear power plant, but also to everyday operation. In the case of leakage through SG tubes to the secondary side of the plant, radiation detection measurements of the secondary flow are taken and calibrated to predict and increase or decrease of leakage through the SG tubes. If excessive leakage occurs, the plant must shut down or the operator must take appropriate action. It is therefore of great interest to not only qualitatively, but quantitatively be able to predict such flow rate with great accuracy.

Most of the recent research related to steam generator tubes is in characterizing defects as well as predicting burst pressures for tubes with defects. There is a plethora of information related to the material behavior under such conditions (Pagan et al, 2009, Kichirka et al, 1997). While some studies do experimentally determine leak rates from steam generator tube defects, they are concentrated on the leak rates after burst. Burst is associated with a very large opening in the side of a tube. The overall prediction of leak rates through cracks is very dependent on the crack opening area, thus the future of leak rate prediction will be the coupling of crack morphology and leak rate models. The work presented here is concentrated on the leak before break, or leak before burst and does not allow for crack growth during leakage.

There is very limited data on the thin wall tube leak rate measurements. Most studies of subcooled choking flow are related to thick wall long tubes with large L/D (Wall thickness/Flow channel length over Hydraulic Diameter) and nozzles (Henry, 1970, Henry and Fauske, 1971). A literature survey performed list the limited sets of data that focus on crack and simulated crack geometries, and can be seen in Table 1, (Wolf and Revankar, 2012, Revankar et al., 2009, 2013). Our survey shows that this geometry has been studied over a large range of pressures and liquid subcoolings however, those data focus on L/D geometries greater than 15. Also, all of the available data have a wall thickness or channel length greater than 10 mm, which is not indicative of thin wall piping or steam generator tubing.

From available tabulated data on crack geometries, the parameter range associated with the current study can further be seen on Figures 1 and 2. From Figure 2, it can be seen that most data points fall at the higher L/D ratios than that of the current study. Also, in the smallest range of L/D very few data exist at lower choking mass flux. From Figure 1, it is clear that of those data points at the smallest L/D, seen in red, no data points fall in subcoolings below 50 °C. In view of this an experimental program of subcooled chocking flow through well characterized specimens of the simulated steam generator tube crack-like geometry is essential.

It is evident from literature (table) that except for the findings of Wolf and Revankar (2012), most research was conducted for crack L/D over 14.7 and wall thickness (flow channel lengths) of over 8 mm. The most recent work by Wolf and Revankar (2012) was conducted on laser cut (therefore rough channel wall) samples with L/D ranging between 4.8 and 6.2 and flow channel length of 3.175 mm for 5 samples and additional testing on a pinhole throughwall geometry. Steam generator tubes thicknesses however range between 1 to 2 mm depending on the material, design and manufacturer and between 12.5 and 19 mm in outside diameter (OD). In addition, L/D ratios are generally on the order of between 0.8 and 2 since the thickness is small but the crack can propagate over time thereby increasing its hydraulic diameter. Thus, further studies are being conducted on cracks of 1.3mm thickness with L/D ratios varying from 1.1 to 2.1

Table 1: Summary of two-phase choked flow studies (Wolf and Revankar, 2012)

Authors	Geometry |fluid	L [mm]	t X w [mm] x [mm]	Dh [mm]	Area [mm^2]	L/D	R roughness [μm]	P [Mpa]	ΔTsub [K]
Agostinelli et al. (1958)	annulus (slit) |steam-water	152-254	(0.15-0.43) X (78.4-79.3)	0.3-0.86	12-38	176-840	(----)	3.5-20.51	10-67
Ryley & Parker (1968)	slits |steam	35.6	(1.27) X (25.4)	2.42	32.3	14.7	(----)	0.008-0.017	0-
Simoneau (1974)	slits |cryogenic N_2	25.4	(25.4)X (0.284-0.3)	0.58	7.42	43.5	(----)	P_{max}=6.8	(----)
Collier et al. (1980)	slits |steam-water	60-75	(0.2-1.12) X (57.2)	0.4-2.2	11.4-64	27-187	0.3-10.2	P_{max}=11.5	33-120
Abdollahian, Levy, Chexal (1983)	cracks |steam-water	18.6-57.2	(0.74-63.5) X (0.0183-1.12)	0.03-1.9	0.015-71.1	30-634	0.3-10.2	3.26-11.53	1-119
Amos & Schrock (1983)	slits |steam-water	63.5	(0.127-0.381) X (14.8-20.5)	0.16-0.77	2.6-7.8	83-400	(----)	4.1-16.2	0-65
Collier et al. (1984)	cracks |steam-water	20	(0.02-0.22) X (0.74-27.9)	0.04-0.44	0.015-6.55	45-500	1.78	P_{max}=11.5	0-72
Kefer et al. (1986)	slits/cracks |steam-water	10-33	(0.097-0.325) X (19-108)	0.26-0.64	5.89-13.93	15-127	20-40	P_{max}=16.0	0-60
John et al. (1987)	slits |steam-water	46	(80)X (0.2-0.6)	0.4	20.0-51.2	115	5-240	4.0-14.0	3-60
Bandyopadhyay et al. (2007)	slits/cracks |steam-water	8	(0.27-0.50) X (15-43.73)	0.54-0.97	5.7-11.8	8.3-14.8	(----)	1.14-8.66	58-264
Wolf & Revankar, (2012)	slit | steam-water	3.175	(0.25-0.50) X (2.4-3.2)	0.55-0.84	0.86-1.92	4.48-6.94	30	6.89	24-46
Samples tested in This Study	slit | steam-water	1.3	(0.83-2.6) X (0.285-0.648)	0.61-1.04	0.513-4.59	1.2-2.1	5-30	6.895	14.1-49.1

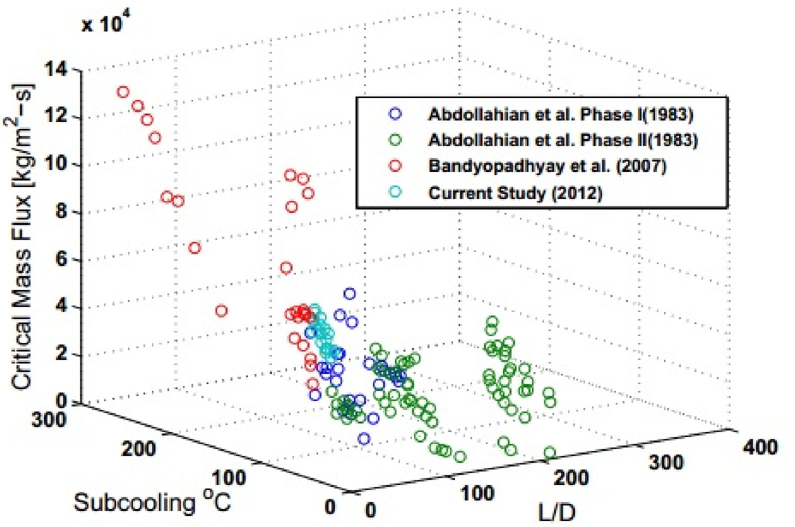


Figure 1: Comparison of current experimental data with relevant data in literature as a function of mass flux, L/D ratio and Subcooling

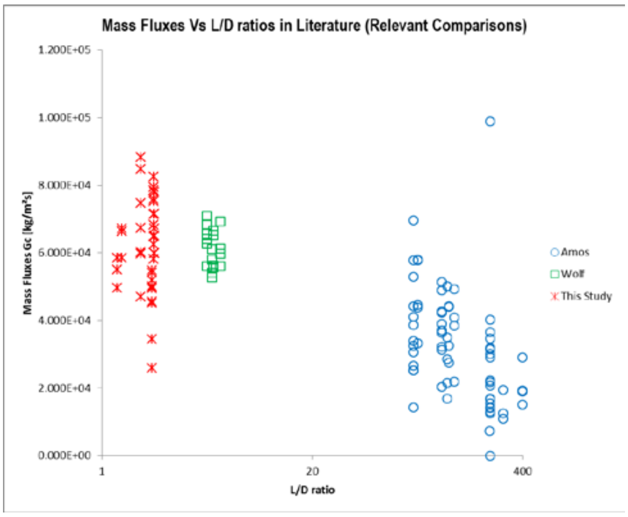


Figure 2: The L/D ratios of this study compared with those the most widely accepted work in literature (Amos and Schrock, 1983)

3. Experimental Test Facility

The purpose of the experimental program was to develop database on critical flow through crack geometries with subcooled liquid flow at the entrance. These databases are then used in validating critical flow models that can be used in assessing leak rates from steam generator tube cracks. With a validated critical flow model and the crack opening area (COA) model, the leak rates from the steam generator tube cracks can be evaluated.

Design of a test facility to measure leak rates of through wall cracks was based on the following goals: (1) The test facility should be modular so that various crack geometries can be studied; (2) The pressure differential across the break should be similar to the prototype operating conditions of about 6.8 MPa (1000 psi), and (3) Facility should be such that tests can be easily repeated. Figure 3 shows a schematic of the test facility design. It consists of a vertical pressure vessel, which serve as the blowdown tank, a water tank where steam condenses and the discharge from crack is collected and measured, a nitrogen supply line to pressurize the vessel with control valve, instrumentation and data acquisition system.

The volume of the vessel was based on the maximum discharge rate expected from the crack. Given this limitation it has been determined that appropriate crack sizes to be studied are between 2 and 8 mm in length. The pressure vessel was first hydrostatic tested at 10.3MPa (1500 psig). Then another hydro test was completed at 14 MPa (2000 psig), which is over 200% of the operating pressure. The vessel has one pressure relief valve (Kunkle brand) with pressure preset to 8.3 MPa (1150 psi) and is mounted with a half inch NPT connection. The pressure vessel is connected via 3/8 inch stainless steel tubing to the compressed nitrogen bottles. Ceramic band heaters are used for heating the pressure vessel contents. Pressure and temperature are measured and a relief valve is placed at the top of the tank. Pressure and temperature are also measured just before the test section. A valve is used to initiate the experiment. As the subcooled water is discharged it flashes and a two-phase critical flow ensues. The discharge steam is condensed by the cooler water bath where the outlet of test section is submerged. The condensed steam and discharged water are collected in the bath volume. This allows for a time averaged mass flow rate to be measured.

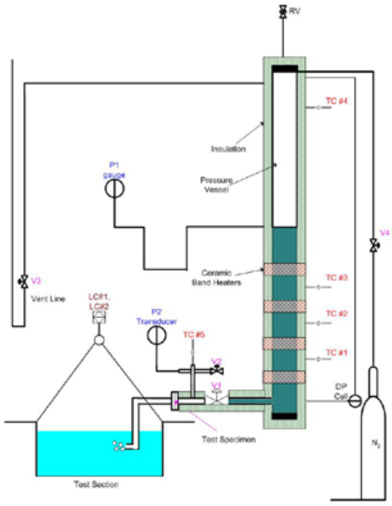
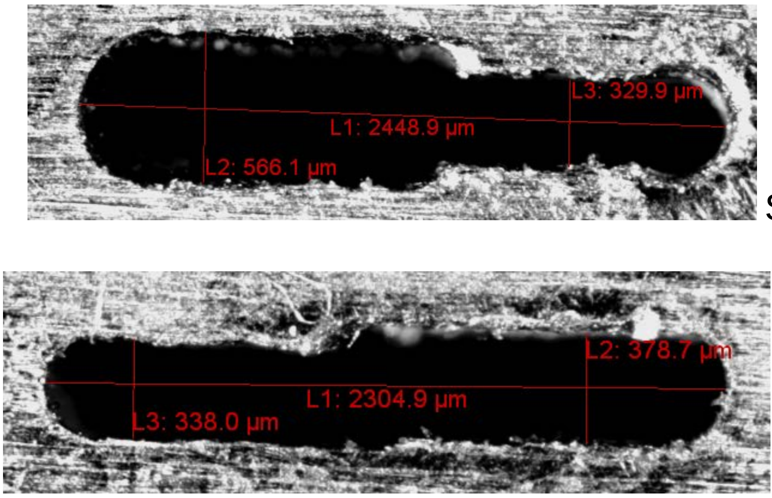


Figure 3: Schematic of test facility

The design of the condensing tank hanging system involves cable supporting the tank with two load cells, which are strain gage type. The tank was suspended using steel wire rope and eyebolts from channel struts spanning the test section support structure. The water level in the vessel is measured with Honeywell DP transducers. The DP cells were calibrated with NIST standard pressure calibrator unit for the range of water level in the pressure tank. The load cells, differential pressure cell and thermocouples were wired to the data acquisition system and were tested. A Labview program was developed for data acquisition and real time data display and monitoring.

3.1 Crack Specimens

Two types of test crack specimens split into two batches were used. Samples were manufactured by laser cutting, drilling and welding semicircular discs together over a stainless steel nipple with the faces milled. This enabled the study of different geometries and surface roughness's. Since slits were cut with laser, the sizes at front (downstream) and back (upstream) side of slit are not same as shown for the specimen #2 in Figure 4.



Slit front

Slit Back

Figure 4: Slit crack test specimen # 2

The effective cross sectional flow area was calculated by averaging the front and back cross section of the slit. Figure 5 shows one of the laser machined and welded slit test specimen with L/D =1.7 to 2.1.

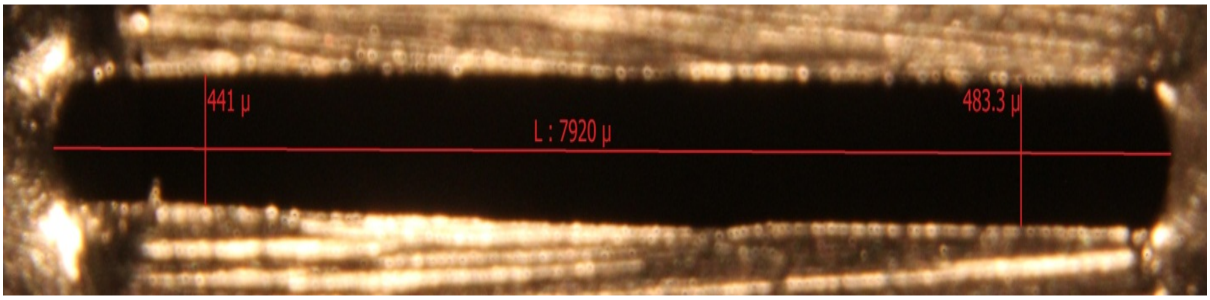


Figure 5: Weld11 with L/D of 1.2

3.2 Subcooled Flashing Discharge Tests

Test of flashing choked flow with heated water were carried out up to a vessel pressure of 6.89 MPa (1000 psi). As the experimental program was designed around testing choking flow through steam generator tube cracks, the most valuable data are those at the highest pressures. The tests carried out at approximately 6.89 MPa, have a pressure differential across the choking plane of near equal value. This is approaching the same pressure differential across the walls of steam generator tubes. However, the differential in actual steam generators is from approximately 14 MPa to 4.5 MPa, not from 7 MPa to 0.1 MPa. Complete scaling of choking flow is not possible within the constraints of this project; however we developed a scaling method.

The tests carried out were varied with subcooling at near the same pressures. Pressures for the tests ranged from 6.87 MPa to 6.60 MPa, with a range of subcooling between 48.1 and 24.7. The highest mass flux for each specimen was obtained at the highest subcoolings, along with the lowest mass flux for the lowest subcoolings as expected. A representation of the mass flux data with respect to subcooling can be seen in Fig. 6. The vertical bars are given as a reference to the error involved in each measurement. As the subcooling increased for each specimen the mass flux increased. In general as the mass flux also increased as the area of the specimens increased. The hydraulic diameters however of specimens #2 and #3 are nearly identical.

The tests carried out for slit #2 at various pressures indicate that the flow rate is dependent on subcooling. Subcooling for the tests varied from 15 C to 29 C. The heated water flashes as it is discharged from the cracks and hence the mass flux discharge decreases with the heated tests. The comparison between cold water tests discharge and heated tests discharge for slit #2 can be seen in Figure 7 as a function of pressure. There is a slight offset in the third data point for flashing flow for slit #2 near 5.3 MPa. Until the test section was taken apart, this lower flow rate than expected could not be explained. It was found that a tiny piece of metal from the steel threads of a recently installed valve had come loose and lodged into the choking area. While studying the flow in small channels this type of interruption can be common, as particles in the stagnant fluid can easily wedge into the flow channel. This leads to the additional step in the test procedure of dismantling the test section before each run, checking and reinstallation.

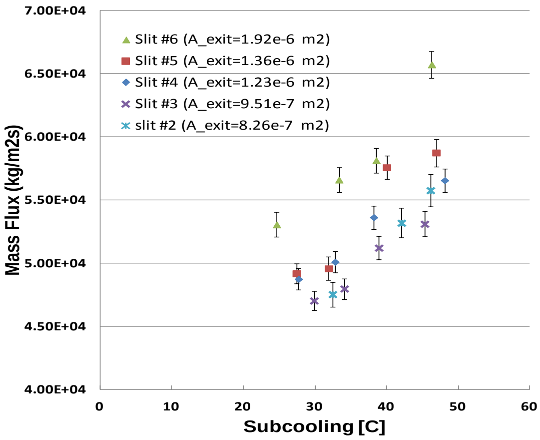


Figure 6. Critical mass flux as a function of subcooling for specimens Slit#2- Slit#6 at about 6.8 MPa

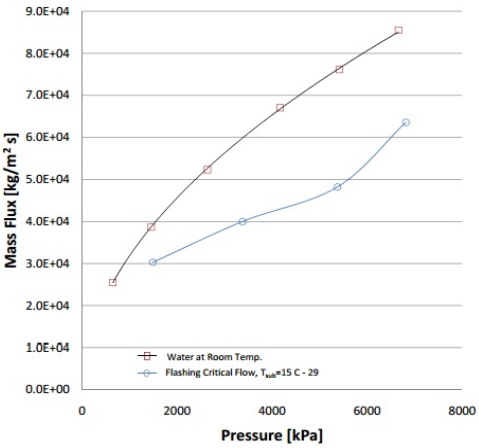


Figure 7: Subcooled Discharge Mass Fluxes compared to Cold Water Discharge for Slit# 2.

Using the test specimens from Batch 2, low subcooling tests were conducted whose flow/channel lengths L/D equaled steam generator tubes, i.e., L/D = 1.2 - 2.1. These tests attempted to simulated CANDU and PWR SG throughwall crack-like conditions as close as possible and the important parameters that affect mass flux are the pressure differential across the crack channel and the degree of subcooling of the fluid. Thus tests were conducted up to maximum pressure differentials of approximately 6.89 MPa (1000 psi) while a variety of subcoolings were studied ranging from 15 °C to 50 °C. Data from these tests were compared with Batch 1 test specimen data in Figure 8, where the choking mass flux data are shown for various subcoolings and pressure conditions. Data shows similar trend as seen from Batch 1 test specimen given in Figure 7, where the choked flow increases with increase in subcooling.

The highest mass flux for each specimen was obtained at the highest subcoolings, along with the lowest mass flux for the lowest subcoolings as expected. As the subcooling increased for each specimen the mass flux increased. In general, as the mass flux also increased as the area of the specimens increased. However, the hydraulic diameters of specimens #2 and #3 are nearly identical. It should be noted that L/D for Slit#2 and Laser8 are 5.2 and 2.1 respectively. When Figures 7 and 8 are compared it shows that the difference between the cold water discharge flow rate and subcooled choking flow rate is lower for smaller L/D test specimen. This indicates that smaller L/D has high thermal non-equilibrium effect on the flashing flow.

Since the L/Ds being studied as a part of this research program are unique, it was decided to take samples from the studies conducted by Amos & Schrock, (1983) whose areas are similar to the samples studied. Studying data for similar pressures and degrees of subcooling will allow us to examine the effect of channel length L and the L/D ratio on mass flux. In Figure 9, the choking mass flux is plotted as function of L/D for various subcoolings. This data clearly demonstrate that increasing subcooling increases mass flux and lowering L/D also increases mass flux.

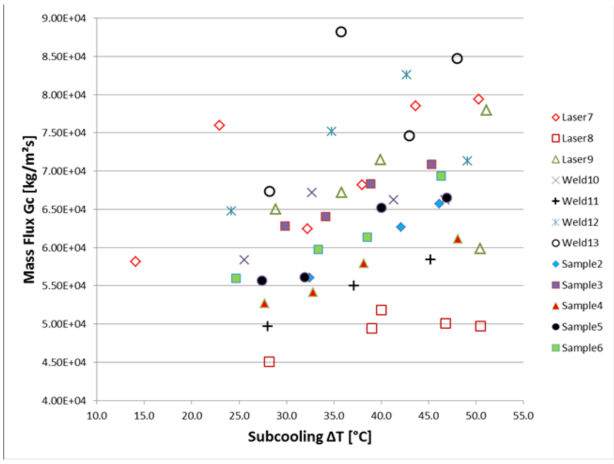


Figure 8 Choking mass flux for test specimen Batch 1 (Slit#2-6 referred as Sample2-6) compared with test specimen Batch 2 (Laser7-9 and Weld 10-13) for different subcooling.

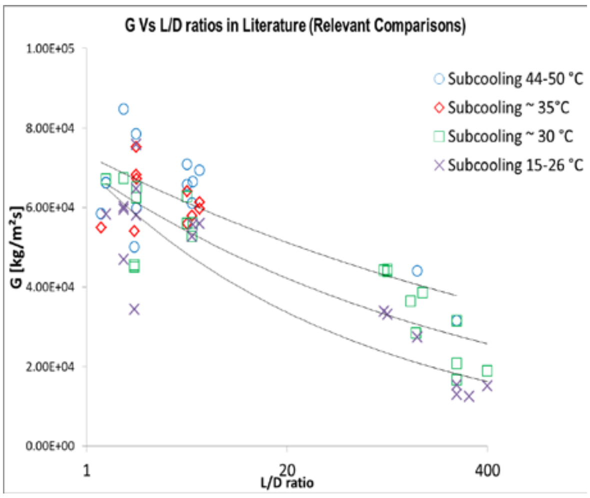


Figure 9 Plot of Mass Flux Vs L/D ratios for similar subcoolings clearly showing the effect of lowering L/D increases mass flux.

4. Analysis of RELAP5 Thermalhydraulics Code

It is of interest to see how system thermal hydraulics codes like RELAP5 and TRACE models fare in modeling choked flow in such geometries and to see if the models break down in anyway when the geometry changes, specifically when the length of the channel decreases.

The choking flow models in RELAP5/MOD3.3 were assessed in two ways. First, the two choking flow models that are available in RELAP were assessed using two experimental runs by Amos and Schrock (1983). The runs were chosen based on the subcooling at two similar pressures of 4.272 MPa and 4.320 MPa. The subcooling for these two runs were 4.8 K and 30.7 K respectively. In the experiment conducted by Amos the channel length was 6.35 cm long and pressure profiles along the channel are available. This allows direct comparison of pressure profiles, and choking flow rate with RELAP5 models. Also, this allows for a starting point to assess how the models behave when the channel length is significantly shortened. The same two models in RELAP were then applied to current data in the case of our test specimens at similar pressures of approximately 3.54 MPa with a range of subcoolings between 19 and 40 K. This analysis, allowed the effect of subcooling at the stagnation condition to be assessed.

As can be seen in Figure 10, at stagnation conditions near saturation (4.8 and 3.9 K subcooling) it is expected that the fluid will flash somewhere along the channel length before the exit (choking) plane. It is evident from the experimental data that much larger pressure drop is seen further downstream of the entrance, indicative of two-phase pressure drop and flashing. The exit pressure observed by Amos was 1909 kPa and 3098 kPa respectively. Both models over predict the exit pressure, and under predict the two-phase pressure drop. The Ransom-Trapp model provides a much better estimate of the single-phase pressure drop when com-pared to the H-F model. Note, that Amos did not actual observe or measure the flashing location in his experiments, it can only be approximately estimated based on the pressure measurements available.

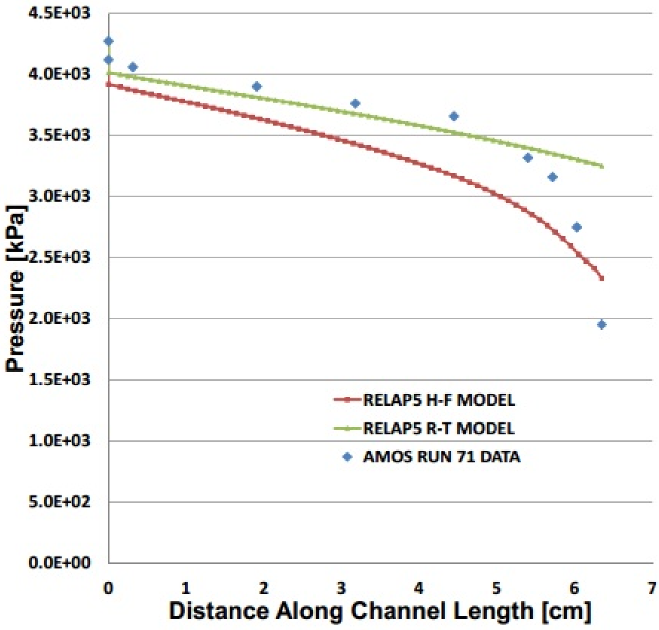


Figure 10: Pressure profile along channel length (L=6.35 cm) for Amos Run#71, with 4.8 K subcooling at 4.272 MPa, with H-F and R-T model predictions.

We observed that the H-F model predicts a much larger void fraction and quality than the R-T model downstream of the flashing point. This could be indicative of the equilibrium flashing criteria at saturation that the H-F model employs. The R-T model on the other hand is using a pressure undershoot which requires a larger drop in pressure to provide the superheat required for bubble growth, therefore the void fraction develops more gradually along the channel length.

5. Conclusion

The ability to estimate the leak rates from the throughwall flaws (cracks, slits, pits, frets, etc.,) is important in terms of radiological source potentially to be released into the environment as well as to the overall safe operation of nuclear power plant. In this study an experimental program and analysis methods were developed to measure and assess the chocking flow rates of initially subcooled water through micro-slits and cracks geometries. Experimental program involved chocking flow for various simulated throughwall flaw geometries for vessel pressures up to 7 MPa with various subcoolings. Measurements were performed on subcooled flashing flow through well-defined throughwall slits and cracks of thin walled piping and vessel components with L/D <5.5, and liquid subcoolings between 14 and 510C. As subcooling increases the flashing critical flow rate also increases. However, from the comparison of data it was noticed that there wasn't a trend that could be spotted of the effect of mass flux on subcoolings, and this appeared to be due to the small channel length, which prevents flow from developing and can essentially be thought of as a stagnant fluid that is accelerated through the crack and choking/flashing slightly downstream of the crack.

Comparisons in between samples of this study and further comparison with other relevant studies in literature revealed:

i. A link between lowering L/D rates and increasing Mass flux for identical conditions of subcooling and pressures. Though this relation is not noticeable for a small change in L/D of 2.5 (from 4.5 to 2), it is easily noticeable over a larger change. There appeared to be a trend which requires further work to develop a correlation.

ii. The reducing importance of subcooling as a parameter affecting subcooled choked flow for small L/Ds and low channel lengths.

iii. The discharge coefficients of the cracks are surprisingly high and not as small as one would expect for an orifice due to the sudden area change.

Both, homogeneous equilibrium and non-equilibrium mechanistic models were developed to model two-phase critical flow through cracks and slits. A comparison of the model results with experimental data shows that homogeneous equilibrium based models significantly under predict critical flow rates in such geometries, while non-equilibrium models improve the accuracy of the predictions. The predictions using homogeneous non-equilibrium model agreed well within 10% of experimental data for chocking discharge flow from such throughwall flaws.

Results of this initial stage of experimental study provided solid foundation to move further and introduce test specimens with real throughwall cracks, developed either by stress corrosion cracking or fatigue induced cracking.

6. References

[1] Alamgir, M., Lienhard, J. (1981), "Correlation of pressure undershoot during hot water depressurization", *Transactions of ASME Journal of Heat Transfer*, 103, 52-73.

[2] Amos, C. N., and Schrock, V. E. (1983). "Critical Discharge of Initially Subcooled Water through Slits", United States Nuclear Regulatory Commission, Division of Accident Evaluation, Office of Nuclear Regulatory Research & Lawrence Berkeley Laboratory, Washington, D.C. 20555: U.S. Nuclear Regulatory Commission NUREG-CR-3475.

[3] Burnell, J., (1946), "The flow of boiling water through nozzles, orifices and pipes", *Journal of Institute of Engineers*, 18, 41-48.

[4] Fauske, H. (1962), "Contribution to the theory of two-phase, one-component critical flow", USDOE, Argonne National Lab, Argonne, Illinois: ANL.

[5] Fauske, H. K., (1985), "Flashing flows or: some practical guidelines for emergency releases", *Process Safety Progress*, 4, (3), 132-134.

[6] Henry, R., (1970), "The two-phase critical discharge of initially saturated or subcooled liquid", *Nuclear Science and Engineering*, 41, 336-342.

[7] Henry, R., Fauske, H., (1971), "The two-phase critical flow of one-component mixtures in nozzles, orifices and short tubes", T*ransaction of ASME Journal of Heat Transfer*, 93, 179-187.

[8] IAEA, (2011), "Assessment and management of ageing of major nuclear power plant components important to safety: steam generators", 2011 Update, International Atomic Energy Agency, Engineering Safety Section, Vienna: IAEA-TECDOC-1668.

[9] Kichirka P.J., M. Blaszkiewicz, W.A. Byers, G. Pierini, (1997), "Oconee unit 2 steam generator tube examination", Electric Power Research Institute report TR-106863.

[10] Majumdar, S., (1999), "Prediction of structural integrity of steam generator tubes under severe accident conditions", *Nuclear Engineering and Design*, 194(1), 31-55.

[11] Moody, F., (1965), "Maximum flow rate of a single component two-phase mixture", *Journal of Heat Transfer*, 87, 134-142.

[12] Pagan, S., Duan, X., (2009), "Characterization and structural integrity tests of ex-service steam generator tubes at Ontario Power Generation", *Nuclear Engineering and Design*, 239, 477-483.

[13] Ransom V.H., Trapp J. A., (1980), "The RELAP5 choked flow model and application to a large scale flow test", Proceedings of the ANS/ASME/NRD

International Topical Meeting on Nuclear Reactor Thermal Hydraulics, NUREG/CP-0014, Vol. 2, p.799.

[14] Ransom V.H., Wang, W., Ishii, M., (1998), "Use of an ideal scaled model for scaling evaluation", *Nuclear Engineering and Design*, 186, 135-148.

[15] Revankar, S.T., Wolf, B., Riznic, J.R., (2013), "Flashing flow of subcooled liquid through small cracks", *Procedia Engineering*, Vol. 56, 454-461 (2013).

[16] Revankar, S. T., Wolf, B., Riznic, J.R., (2012), "Metal fatigue crack growth models", *International Journal of Advanced Engineering Applications*, Vol.1, Iss.4, pp.85-91.

[17] Revankar, S.T., Riznic, J. R., (2009), "Assessment of steam generator tube flaw size and leak rate models", *Nuclear Technology*, 167, (1), 157-168.

[18] Ru X, Staehle, R. W., (2013), "Historical experience providing bases for predicting corrosion and stress corrosion in emerging supercritical water nuclear technology: Part 1" – Review, Corrosion, 69, (3), 211-229.

[19] Schrock, V., (1992), "Critical flashing flow in pipes and cracks", *Dynamics of Two-Phase Flows*, 571.

[20] Schrock, V., Revankar, S., Lee, S., Wang, C., (1982), "Critical flow through intergranular stress corrosion crack in pipes", U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Washington, D.C. 20555: Lawrence Berkeley Laboratory.

[21] Staehle R.W., Gorman J.A., (2003), "Quantitative assessment of submodes of stress corrosion cracking on the secondary side of steam generator tubing in pressurized water reactors: part 1", Corrosion, 59, (11), 931-994.

[22] Trapp, J.A., Ransom V.H., (1982), "A choked-flow calculation criterion for nonhomogeneous, nonequilibrium, two-phase flow", *Int. J. Multiphase Flow*, 8, (6), 669-681.

[23] Vadlamani, R.A., Revankar S.T., (2013), "An experimental study of subcooled choked flow through steam generator tube cracks", Purdue University report NU/NE-13-15.

[24] Wallis, G. B., (1969), "One-dimensional two-phase flow", McGraw-Hill.

[25] Wallis, G. B., (1980), "Critical two-phase flow", *International Journal of Multiphase Flow*, 6, 97-112.

[26] Wolf, B., Revankar, S. T., (2012), "Subcooled choked flow through steam generator tube cracks", Purdue University Report, PU/NE-12-09.