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An experimental analysis of subcooled leakage flow through slits from high pressure high temperature pipelines

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ABSTRACT

The work presented here is an experimental investigation of the critical flashing flow of initially subcooled water through circumferential slits in pipes. The study provides first hand information about the prediction of leak flow rates in piping and pressure vessels retaining high temperature and high pressure. The dedicated experimental facility loop simulates the thermal hydraulic condition of Pressurized Heavy Water Reactors (PHWR). The critical flow characteristics found for varying leakage cross sections at different stagnation pressure and different degree of subcooling has been demonstrated in this paper. A marked decrease in mass flux has been found as subcooling decreases for a fixed stagnation pressure. More observation has revealed that the tighter slits or openings with very short duct as small as 0.8 cm flow length have different flow behavior than greater opening dimensions or with longer flow channels or that for nozzles. The critical flow has been seen to occur at higher pressure differentials along the flaws and prominent changes in the flow rate is reported to occur with varying dimensional parameters of the slit or cracks.

1. Introduction

Leak Before Break analysis has been proven a mandatory safety analysis for cost-effective high pressure high temperature system design. This particular analysis looks for formation of cracks and their subsequent growth before complete disruption of the whole system. Safety is a prime concern in the nuclear reactors for their primary system pipelines carrying radioactive coolants. Therefore these pipelines are designed with high grade tough (ductile) materials which are least susceptible to instantaneous breakage. An example of that kind is a Double Ended Guillotine Break (DEGB) assumption which necessitates building costly shields against dynamic effects like potential missile, pipe whipping, blowdown jets etc. On the other hand analysis of fracture mechanism and rupture dynamics shows possibility of a stable and detectable leak flow occurring through the leakage. These cracks grow at a stable rate due to the sustaining mechanical and thermal loads. Therefore leak detection and measurements are primary requisite in LBB analysis which also provides useful information about the probable crack size and geometry. In this context works by Bertholome et al. [1] on leak detection system have demonstrated that certain pipes under specific thermal hydraulic conditions produces small through wall cracks resulting in coolant leak rates in the range of 0.05 kg/s which are easily detectable by installed leak detection system. Once the leak flow or loss of coolant (LOCA) exceeds a certain limit called reactor shutdown limit, operation is halted and corrective measures are taken for that particular section. Applying US Nuclear Regulatory Commission's factor of safety of 10, the leak detection capability or reactor shutdown limit is set to 0.5 kg/s. Thus LBB approach results in significant saving in plant layout, labor cost and radiation dosage for maintenance. Analyzing the flow pattern and upstream–downstream conditions for this kind of leakages a critical flow is suspected with instant flashing. While passing through the leakage at local acoustic velocity the flow gets choked and become independent of further change in pressure differential. Rapid transformation from subcooled single phase stagnation condition to a two phase accelerated flow into very low pressure regions, lacks thermodynamic equilibrium due to very small residence time. A flow problem of this kind in a compressible zone involves a lot of unknown parameters which makes it more unpredictable. An experimental approach in this context becomes very useful to have direct data about the condition. For larger confidence on the LBB methodology rigorous experimentation at reactor conditions and validation of critical flow models such as those developed by Henry and Fauske [2] are

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Nomenclature

G	Mass Flux
c	Critical quantities
COA	Crack/slit opening area
L	slit depth/pipe thickness
BC	Buffer Chamber

extremely useful. Determination of critical leak flow at different reactor conditions and its change with thermal hydraulic transients is therefore an important issue to be taken care of.

Experiments directly related to the problem of leak flow through simulating artificial slits are rare. In a few of them, Agostinelli et al. [3] performed tests in 1957 with annular constant area passages of hydraulic diameter 0.15–0.43 mm. The pressure range was 3.5–20.5 MPa and subcooling from 9.3 °C to 67 °C. Collier et al. [4] investigated critical flow of water in rectangular slits. Flow rates were much smaller than had been anticipated. Fauske [5] has measured mass flux (mass flow per unit area) and pressure profiles for the two phase critical flow of saturated steam water mixture through pipes but couldn't found any observable dependence of flow rate on test section diameters. Henry et al. [6] investigated critical flow and measured void fraction and pressure profiles and noted the decrease of flow rate with increase in L/D ratio. Most of the studies on subcooled water stagnation conditions have been carried out in nozzles or short ducts. Shrock et al. [7] reported the results for initially subcooled water, including steady-state flow studies and pressure distribution in the nozzle ranging between 200 and 890 Bar. Study by Zimmer et al. [8] provides detailed information about the flashing flow of subcooled water through convergent divergent nozzles. Shiba and Curet [9] investigated flow through large diameter ducts (15–103 mm) at typical PWR pressures of 15.7 MPa. Sozzi and Sutherland [10] also studied blowdown phenomenon through short pipes at lower subcooling with initial pressure of 6.8 MPa. Most of these relevant works have been summarized by Hall [11] and Abdollahian et al. [12] on 1982. More recently on 1984 Amos & Shrock [12] have investigated critical flashing flow of initially subcooled water through rectangular slits experimentally as well as theoretically. Their measurements of critical mass flux and critical pressure provides considerable amount of data in the context of present work. The present work describes experimentation with circumferential slits in pipes of varying geometries with dedicated experimental facility which simulates the thermal hydraulic condition of a PHWR. The findings reported could be beneficial for modern pipeline designs and safety concerns. It further provides knowledge base about the timing of actions to prevent a small break Loss of Coolant Accident (LOCA), operation of safety relief valves to avoid accidents like that occurred at the Three Mile Island plant in March 1979.

2. Leak test facility

To maintain the required stagnation thermal hydraulic parameters and satisfy a steady state condition, was the primary requisite for proper simulation. The facility was developed with an aim to maintain the desired reactor pressure and temperature at the slit upstream for a sustained period ($\approx$10 min) during which the flow stabilizes to a steady flow through the slit pipe. Inside the High Pressure and High Temperature (HPHT) loop a Buffer Chamber (BC) of 1.25 m^3 capacity (0.6 m dia. Sch. 100 and 5.5 m height) connected with a nitrogen manifold at its top, was used as a source of pressurized subcooled water. The nitrogen manifold was connected with 48 nitrogen cylinders (rated 140 bar) through a control valve (PCVN) and headers from the nitrogen room. Nitrogen released by the valve occupies the top area of BC and pressurizes the water from a predefined water level inside BC. To ascertain a nonflashing situation pressure was increased according to the temperature rise at the time of heating. The huge number of cylinders was necessary to maintain certain upstream pressure at the time of blowdown or leakage happening downstream. Use of nitrogen as a cover gas for subcooled water introduces the possibility of dissolution of nitrogen as postulated by several researchers like Edwards [15]. Henry et al. [6] used nitrogen gas to maintain a constant stagnation pressure too. The present data also involves influences of nitrogen mostly at higher temperatures. The level of water in the BC was a primary thing to note because the pressurization process of water involved a swelling effect inside BC. Therefore a differential pressure transducer was devised with a level transmitter to keep a watch on the water level. Also the level indicator was digitally interlocked with the reciprocating pump (P1) to fill up the BC to certain level. At the time of leak test experiment, level-switch provided important information about the current inventory of water inside BC. Fig. 1 provides a schematic of the test facility with the key components identified.

Experiments were done in batch mode. No direct recirculation was involved to pressurize the flow through leak. Whatever leak flow was obtained was due to the stagnation pressure at the upstream. Water inside the BC was heated by Thermic Fluid Heating System (TFHS) which runs around the BC through a helical path in a jacket. Heat to the thermic fluid (Therminol 59, boiling point 350 °C) was provided by burning high speed diesel in a burner, aided with circulating pump and blower. The heating system is designed for 5 MPa pressure and 350 °C temperatures. A desired temperature inside the BC is maintained with heating control mechanism. The temperature obtained at BC was found by installing two RTDs (range 30 °C–350 °C) at the bottom of the BC connected with temperature transmitters. These data was really useful to observe the huge thermal inertia of the whole system and to control the thermic fluid circulation around BC at proper timing. The HPHT loop was capable of containing subcooled water at an average thermal-hydraulic condition of PHWR (90 bar, 250 °C) during the experimental duration. The heated subcooled water found its way into the test section through a coriolis type mass flow meter (range 0.01–0.8 kg/s). Pressure and temperature transmitters were installed upstream to the test section to measure the

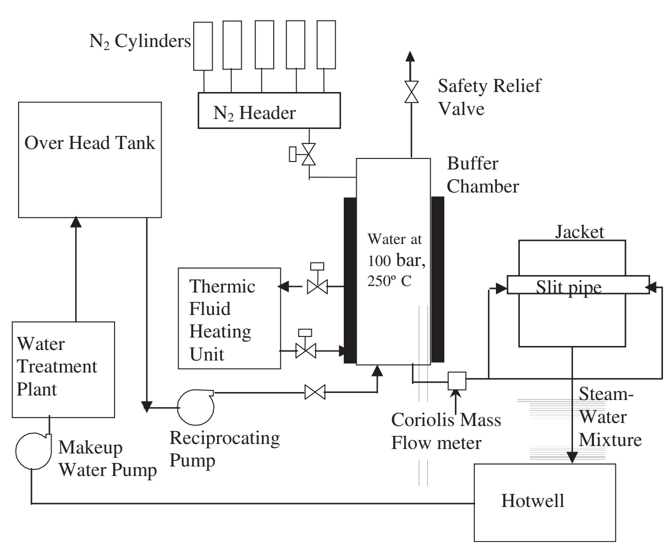

Fig. 1. Schematic of the test facility.