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An Experimental Investigation on the behavior of Pressure Tube under Symmetrical and Asymmetrical Heating Conditions in an Indian PHWR

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***Abstract*—**Thermal behavior of fuel channel under loss of coolant accident (LOCA) is a major concern for nuclear reactor safety. LOCA along with failure of emergency cooling water system (ECC) may leads to mechanical deformations like sagging and ballooning. In order to understand the phenomenon an experiment has been carried out using 19 pin fuel element simulator. Main purpose of the experiment was to trace temperature profiles over the pressure tube, calandria tube and clad tubes of Indian Pressurized Heavy Water Reactor (IPHWR) under symmetrical and asymmetrical heat-up conditions. For simulating the fully voided scenario, symmetrical heating of pressure was carried out by injecting 13.2 KW (2 % of nominal power) to all the 19 pins and the temperatures of pressure tube, calandria tube and clad tubes were measured. During symmetrical heating the sagging of fuel channel was initiated at 460 ^{o}C and the highest temperature attained by PT was 650 ^{o}C. The decay heat from clad tubes was dissipated to moderator mainly by radiation and natural convection. The highest temperature of 680 ^{o}C was observed over the outer ring of clad tubes of fuel simulator. Again, to simulate partially voided condition, asymmetrical heating of pressure was carried out by supplying 8.0 kW power to upper 8 pins of fuel simulator and temperature profiles were measured. Along the circumference of pressure tube (PT) the highest temperature difference of 320 ^{o}C was observed, which highlights the magnitude of thermal stresses under partially voided conditions.

***Keywords*—**LOCA, ECCS, PHWR, Ballooning, channel heat-up, pressure tube, calandria tube

I. INTRODUCTION

THE first phase of Indian nuclear power reactor was focused on construction of 220 MWe and 540 MWe ratings across various parts of the country. A schematic diagram of primary heat transportation system in PHWR is shown in Fig. 1. The coolant flows through 306 horizontal channels which are housed in calandria vessel and submerged in heavy water called moderator. The coolant flows through half of the channels in one direction and in remaining channels in opposite direction.

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The carbon dioxide gas at atmospheric pressure is filled in gap between calandria and pressure tube for thermal insulation. The nuclear heat is removed from fuel bundles by heavy water coolant in primary circuit and transferred to secondary circuit in steam generators. Steam produced in secondary circuit is then supplied to turbines for power generation. During normal operations, the creep sagging of pressure tube is common problem for horizontal tube type reactors. Fast or slow heat-up causes PT to deform either by ballooning or sagging depending on its internal pressure. The failure of pump discharge line, reactor inlet header, single feeder pipe failure etc. come under single failure event. LOCA along with failure of ECCS or LOCA along with failure of ECCS and moderator cooling system come under multiple failure events. Under certain accident conditions, both single and multiple fails and it become difficult for nuclear reactors to remove decay heat from the core because fuel bundles are uncovered due to lack of water. This scenario that has a probability of about 10^{-7} per year is a severe accident (SA) which leads to fuel and PT heat-up as well as its deformation. Brown et al. [1] analyzed the PT deformation in CANDU reactors for a large LOCA with a loss of emergency core coolant injection. The pressure tube integrity was assessed by Gulsani [3] for a CANDU reactor channel experiencing a small LOCA coincident with total loss of ECCS. Kohn et al. [5] analysed if the pressure is still high enough in an early heated channel, the pressure tube can balloon uniformly and contact the calandria tube, establishing an effective heat transfer path to the moderator. During the study of PT ballooning by creep, Shewfelt et al. [9] observed that a rise in temperature above 450 ^{o}C would produce rapid creep deformation in the pressure tube and internal pressure generates large hoop stress deforming it plastically outwards. Shewfelt et al. [10] experimentally studied the longitudinal creep behavior of Zircaloy at relatively high temperature of 650–950 ^{o}C. At this high temperature, the strain caused by the fuel bundle weight and self-weight of the pressure tube is enough to cause sagging at the unsupported region. A series of experiments was carried out on pressure tubes by Shewfelt et al. [9] and Gillespie et al. [2] to estimate the ballooning and sagging behavior separately. Yuen et al. [11] experimentally analyzed the structural integrity of PT and assessment of moderator as a heat sink has been carried out for CANDU reactor for circumferentially symmetric and asymmetric heat-up conditions. Gupta et al. [4] observed that during postulated low frequency events like LOCA along with the failure of ECCS, the cooling environment for the bundles degrade and this results in heat-up of the fuel bundles which in turn heats up the PT through radiation heat transfer.

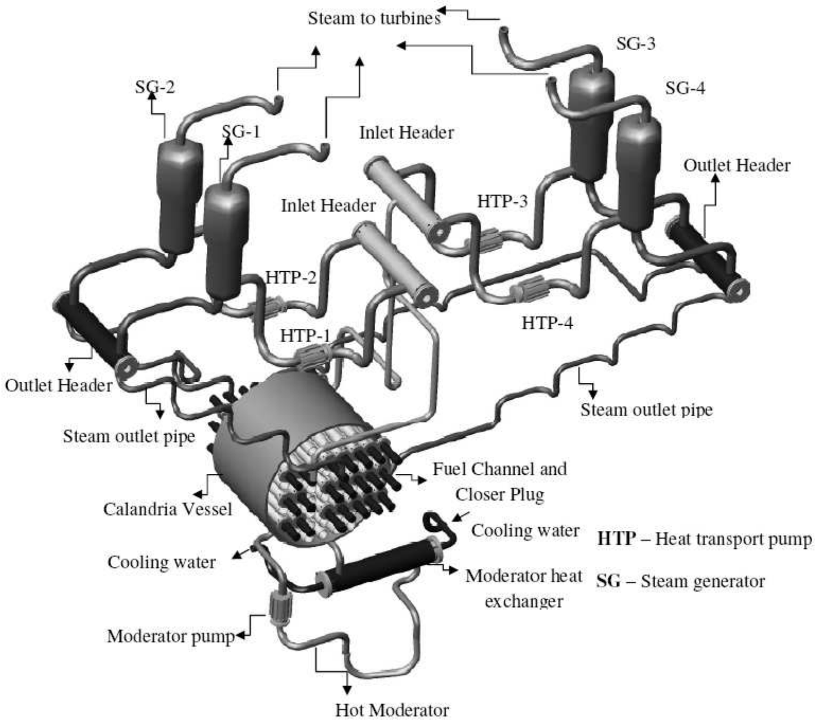

Fig. 1 Schematic diagram of heat transportation system

The prediction of creep deformation of PT used in Indian PHWR under simultaneous internal pressure and weight has been reported by Majumdar et al. [7]. Combined sagging and ballooning of PT under LOCA at different heat-up rates has been carried out by Nandan et al. [8]. In order to trace temperature profile over pressure tube of IPHWR under full voided and partially voided conditions, an experiment has been carried out using a 19 pin fuel element simulator.

II. Experimental Set-up and Procedure

Fig. 2 shows the schematic diagram of experimental set-up. The set-up consists of a mild steel tank of 1000 mm × 500 mm × 500 mm size with 5 mm sheet thickness. The calandria tube (CT) having 1000 mm length was fixed in the tank and the joint with the tank wall was made leak proof.

A 1500 mm long PT was used as test section, middle 1000 mm was inside the CT and remaining 250 mm length was outside of the CT on each side. The PT was supported on two vertical stands and was made concentric with CT with the help of a hand screw arrangement with a vernier scale. One end of PT was made fixed end and another was made a free end in order to attain the situation close to that in the reactor.

A 19 pin fuel simulator of IPHWR with its assembly in pressure tube is shown in Fig. 3. The major components of fuel simulator were the heater rod assembly, spacer, current distribution disc and copper rod for supply of power. The fuel simulator was designed for a maximum heating capacity of 17.5 KW. Total input power was distributed among outer, middle and centre rod in ratio of 1.4: 1.1: 1 respectively. The clad tubes were insulated from heating rod by compacted castable Alumina (Al_2O_3). Two identical spacers were located at the both ends to hold the 19 pins firmly in desired positions. Two copper discs of 10 mm thickness were located beside spacers at both ends to distribute power among all the heating rods (Fig. 4). The couplers were located among the heating rods to supply power as per requirement. To create asymmetrical heating conditions upper 8 rods were coupled and remaining was decoupled. The temperature of PT was measured with mineral insulated ungrounded K-type thermocouples of 0.5 mm outer diameter while J-type thermocouples of 1.0 mm outer diameter were used for the temperature measurement of CT. All the thermocouples were calibrated prior to installation. The thermocouples were located at five positions (±20 cm, ±40 cm and at centre) of pressure tube and calandria tube. They were fixed over outer surface of PT and CT with help of 8mm×4mm×0.1mm zircalloy foils. The tip of thermocouples were inserted in small groves (18 mm ×.1mm×.1mm) over outer surface of tubes and then covered by zircalloy foils. The foils were spot welded over the surface of PT and CT. Again to gather temperature profiles over clad tubes law of symmetry was adopted and thermocouples were located at 20 cm apart from centre of fuel simulator on both sides. The details of thermocouple locations on PT, CT and clad tubes are shown in Fig. 5. All the thermocouples and transducers were calibrated prior to installation and were connected to the Data Acquisition System (DAS).

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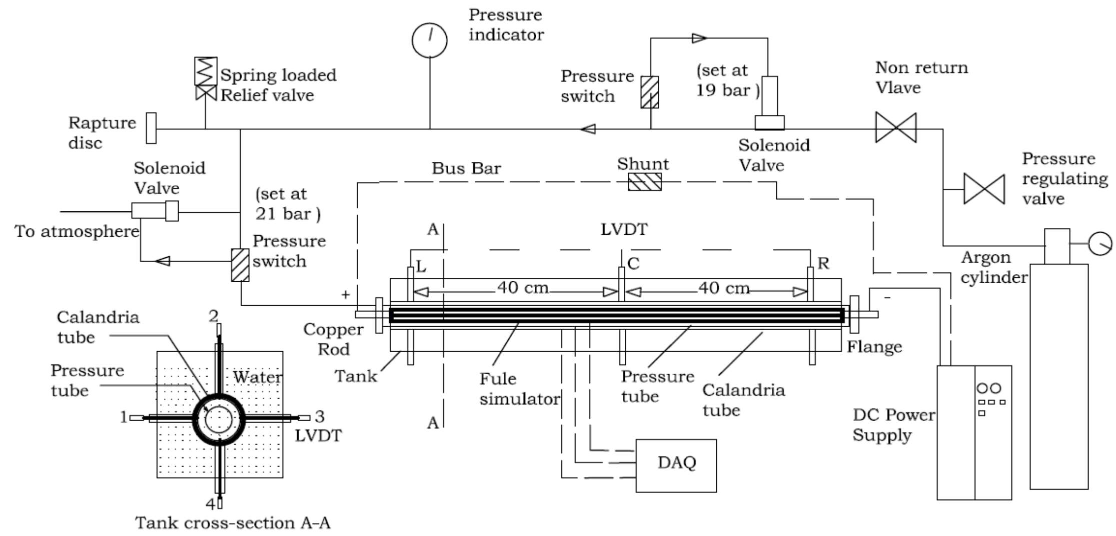
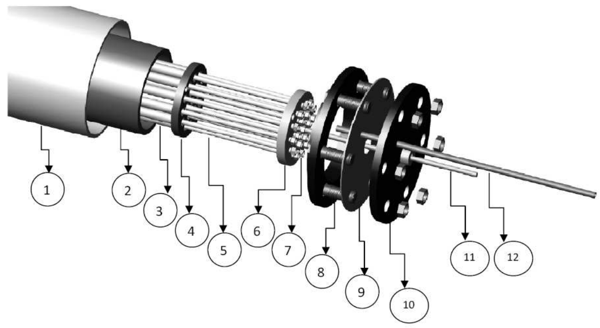


Fig. 2 Schematic diagram of experimental set-up



1. Calandria tube, 2. Pressure tube, 3. Clad tube, 4. Spacer, 5. Heating rod, 6. Current distribution disc, 7. Coupling nuts, 8. Flange hub, 9. Packing plate, 10. Cover plate. 11. Copper rod

Fig. 3 Details of Fuel rod simulator

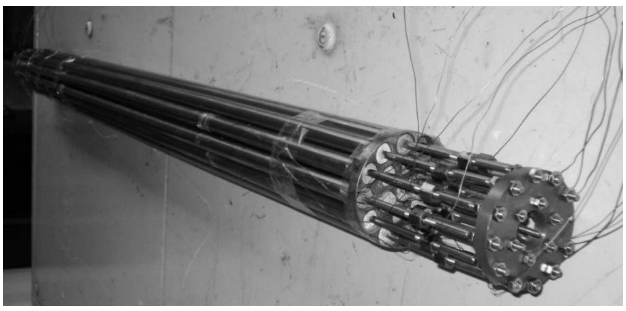


Fig. 4 Photograph of fuel rod bundle

First of all, the tap water was filled in the tank up to 400 mm height from the base of the tank submerging the CT. The water in the tank was heated to a temperature of 60^{o}C using immersion heater, followed by the heating of PT using rectifier.

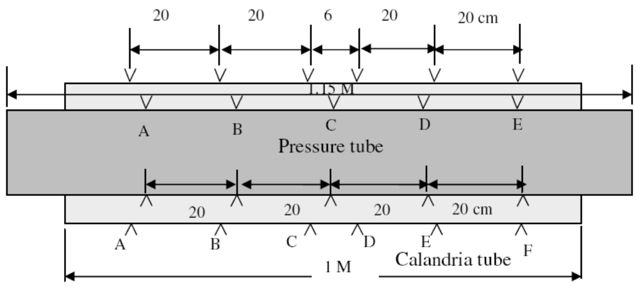


Fig. 5 Location of Thermocouples

The step input power of 13.5 kW was given to the test-section by injecting 3500A current in the tube bundle. The experiment was continued till the PT stopped sagging. The corresponding temperature and displacement were recorded during the process at a time interval of 0.1 seconds. The similar procedure was adopted, at the heating rate of 8.0 kW, when only top eight tubes inside PT were activated to study the asymmetric heating of PT.

III. RESULTS AND DISCUSSION

Under loss of coolant accident heat is rejected to moderator in calandria vessel is mainly due to radiation and convection heat transfer. A symmetrical heating of PT was carried out under unpressurised state and circumferential temperatures of PT and CT were measured at five different locations by supplying a ramp power of 13.2 KW and then average of these temperatures has been plotted against time (Fig. 6). It has been observed that temperatures at upper periphery of PT at locations P2 and P3 were higher as compared to other locations along the circumference. This is in well agreement with findings of Kuehn and Goldstein [6] for concentric cylinders. They predicted that local equivalent conductivity at inner cylinder wall increases from top to bottom and due to this reason heat transfer rate at bottom is higher as compared to top of PT circumference. Significant decrement in slope of temperature rise at P5 has been observed after 470 ^{o}C. This is due to sagging of pressure tube which leads to increment in heat transfer from PT to CT and is in well agreement with Shewfelt et al. [9] and Nandan et al. [8].

Similarly as shown in Fig. 6(b) temperatures at C4 and C5 positions on calandria tube were significantly high. This is due to sagging of clad tubes and pressure tube which leads to increase in heat transfer rate from PT to CT at bottom. The Power in the fuel rod bundle was distributed among the outer, middle and centre rods in ratio 1.4:1.1:1 respectively. The variation of temperature among the clad tubes under 13.5 KW ramp power is shown in Fig. 6(c). Maximum temperature has been observed on outer ring clad tubes and minimum at centre of fuel simulator. There was a significant decrement in slop after 1300 seconds due to sagging of pressure tube. Sagging of pressure tube tends to increase heat transfer rate from its bottom periphery and hence temperature rise rate declines. After 2000 seconds from start-up of experiment, the temperature of clad tubes has been stabilised because of heat balance between its generation and dissipation to moderator in tank.

To simulate partially voided conditions, asymmetric heating has been carried out by supplying power of 8.0 kW to only upper eight heating rods as shown in Fig. 7. The temperature profiles have been measured at five different locations and the average temperature was plotted against time. Fig. 8 shows temperature at upper periphery of pressure tube (P5 & P4) was significantly high as compared to the bottom. The highest temperature difference of 320 ^{o}C has been observed between upper and lower periphery of PT (P5 & P2 respectively) after 450 seconds. After 660 seconds of ramp input the temperature at upper periphery stabilized because of the balance between heat generation and dissipation. The temperature at other positions over circumference of PT still increased because of convection currents and conduction from upper periphery to lower positions.

Fig. 9 shows that highest temperature difference of 550^{o}C has been observed across top and bottom clad tubes (HO1 & CO1) after passage of 300 seconds of heating. Again there is steep decrement in slop of temperature after 320 seconds because of rise in heat dissipation to bottom side.

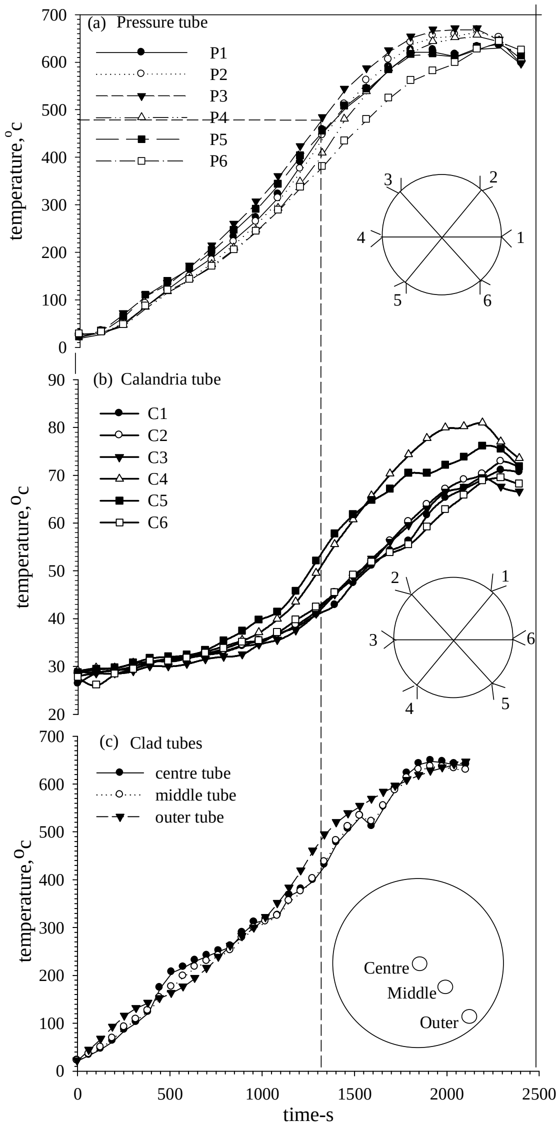


Fig. 6 Variation of the temperature PT, CT & Clad tubes under symmetrical heating

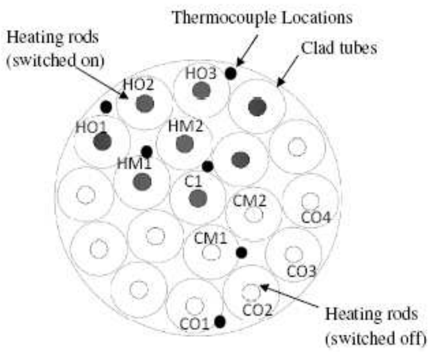


Fig. 7 Location of activated rods of fuel rod simulators

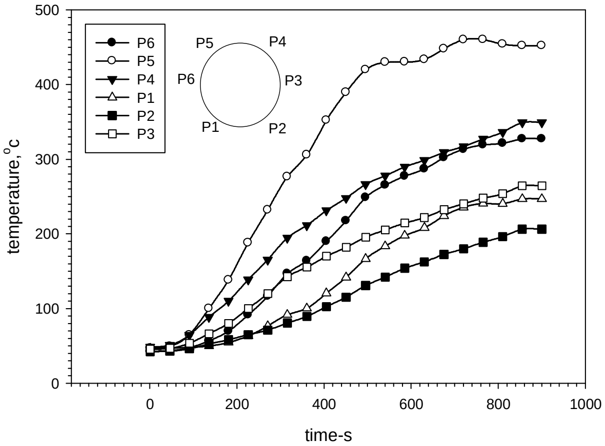


Fig. 8 Temperature transient over PT under asymmetrical heating

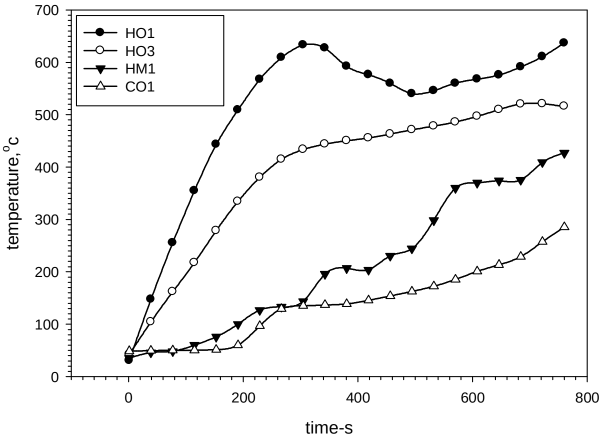


Fig. 9 Temperature over clad tubes asymmetrical heating

IV. Conclusion

1. When a power of 13.2 KW was supplied to 19 pin fuel simulator, sagging of pressure tube has been observed at 460 ^{o}C which agrees well with Shewfelt et al. [9] and Nandan et al. [8].
2. The highest temperature has been observed at bottom of pressure tube (P4 and P5) due to sagging of clad tubes.
3. The power has been distributed among inner, middle and outer rings in ratio of 1:1.1:1.4 respectively. Accordingly the highest temperature has been observed at outer clad tubes and the lowest temperature at inner clad tube.
4. The highest temperature difference of 320 ^{o}C has been observed across the circumference of PT during the asymmetric heating.
5. Under asymmetrical heating the highest temperature difference of 550 ^{o}C has been observed across top and bottom clad tubes.

Acknowledgements

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References

[1] R.A. Brown, "*Degraded cooling in a CANDU reactor*," Nucl Sci Eng 88(3), 1984, pp. 425–35.

[2] G.E. Gillespie, R.G Moyer, D.G. Litke,"*The experimental determination of circumferential temperature distributions developed in pressure tube during slow coolant boil down*," Proc. CNS 8th Annual Conference, Saint John,1987, pp. 241-248.

[3] P.Gulshani, "*Prediction of pressure tube integrity for a small LOCA and total loss of emergency coolant injection in CANDU*," Trans Am Nucl Soc 55(Nov), 1987, pp-461.

[4] S.K. Gupta, B.K. Dutta, V. Venkatraj, A. Kakodkar, "*A study of Indian PHWR reactor channel under prolonged deteriorated flow conditions.*," IAEA TCM: Advances in Heavy Water Reactor, Bhabha Atomic research centre, India, 1996.

[5] E. Kohn, G.I. Hadaller, R.M. Sawala, G.H. Archinoff, S.L. Wadsworth, "*CANDU fuel development during severely degraded cooling: experimental results.*," In: Canadian Nuclear Society Conference, Ottawa, Ontario, 1985.

[6] T.H. Kuehn, R.J. Goldstein, "*An experimental and theoretical study of natural convection in the horizontal annulus between horizontal concentric cylinders.*," Journal of Fluid mechanics 74 (4), pp- 695–719.

[7] P. Majumdar, D. Mukhopadhyay, S.K. Gupta, H.S. Kushwaha, V. Venkat Raj,"*Simulation of pressure tube deformation during high temperature transients*," International Journal of Pressure Vessels and Piping 81 (7), 2004, pp-575–581.

[8] G. Nandan, P.K. Sahoo, R. Kumar, B. Chatterjee, D. Mukhopadhyay, H.G. Lele, "*Experimental investigation of sagging of a completely voided pressure tube of Indian PHWR under heatup condition*," Nuclear Engineering and Design 240 (10), 2010, pp- 3504-3512.

[9] R.S.W. Shewfelt, L.W. Layall, D. P. Godin, "*High temperature creep model for Zr-2.5 wt.% Nb pressure tubes*," Journal of Nuclear Materials 125, 1984, pp- 228–235.

[10] R.S.W. Shewfelt, L.W. Layall, "*A high temperature longitudinal strain rate equation for Zr–2.5 wt% Nb pressure tubes*," Journal of Nuclear Materials 132, 1985, pp- 41–6.

[11] P.S. Yuen, C.B. So, R.G. Moyer, D.G. Litke, "*The experimental measurement of circumferential temperature distributions developed on pressure tubes under stratified two-phase of conditions*," In Proc. CNS 9th Annual Conference, Winnipeg, Manitoba, 1988,pp-120-126.