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Flow boiling heat transfer in a helically coiled steam generator for nuclear power applications

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ABSTRACT

Forced convection boiling of water was experimentally investigated in a 24 m long full-scale helically coiled steam generator tube, prototypical of the steam generators with in-tube boiling used in small modular nuclear reactor systems. Overall, 1575 axially local and peripherally averaged heat transfer coefficient measurements were taken, covering operating pressures in the range of 2–6 MPa, mass fluxes from 200 to 800 kg m^{-2} s^{-1} and heat fluxes from 40 to 230 kW m^{-2}. The heat transfer coefficient was found to depend on the mass flux and on the heat flux, indicating that both nucleate boiling and convection are contributing to the heat transfer process. Seven widely quoted flow boiling correlations for straight tubes fitted the present helical coil databank with a mean absolute percentage error within 15–20%, which was comparable with the experimental uncertainty of the measured heat transfer coefficient values, thus indicating that curvature effects on flow boiling are small and negligible in practical applications.

1. Introduction

Helically coiled tubes have many advantages over conventional straight tubes that make them extensively used in steam generators and heat transfer equipment in chemical plants, refrigeration systems, power stations and nuclear reactors. For example, they provide a greater heat transfer area per unit volume, which make them attractive in applications where compactness is required, such as marine propulsion and integral layout nuclear reactors. In addition, helically coiled tubes are less susceptible to problems associated with thermal expansion, which can present operational constraints on straight tube systems operating at high temperature. Moreover, the large radial accelerations induced by the helical path maintain the tube internal surface wet to higher vapor qualities than in a straight pipe, thus delaying dryout and the associated degradation in the heat transfer coefficient, and also improve the heat transfer effectiveness after dryout promoting de-entrainment of the liquid droplets from the vapor. Recently, helically coiled steam generators are receiving renewed interest, promoted by their frequent use in advanced small modular nuclear reactor systems. According to the International Atomic Energy Agency [1], nuclear reactors are classified as small modular reactors when their equivalent electric power output is less than 300 MW. A recent review [2] identified more than 45 different small modular reactor designs in various stages of development, including advanced water cooled reactors, high temperature gas cooled reactors, liquid metal cooled reactors and molten salt reactors. The modular approach of advanced small modular reactor systems offers many advantages over traditional nuclear power plants, including lower initial capital investment, simplified design, economy of mass production, enhanced passive safety and proliferation resistant features [3]. In addition, the modular concept provides greater flexibility, enabling small modular reactor systems to meet much smaller local demands for power [4]. The most mature small modular reactor designs with the potential for near- and mid-term commercial deployment are the integral pressurized water reactor designs, which house all the components usually associated with the primary circuit of a pressurized water reactor within a single reactor vessel. They are considered to pose the lowest technical risk, since they combine proven technologies and many years of commercial operating experience with traditional pressurized water reactors, with a range of novel

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Nomenclature

d	tube diameter (m)	Pr_w	Prandtl number evaluated at inner tube wall temperature (–)
D	coil diameter (m)	Re	Reynolds number (–)
f	Fanning friction factor (–)	μ	fluid viscosity at fluid bulk temperature (kg m^{-1} s^{-1})
Nu	Nusselt number (–)	μ_w	fluid viscosity at inner tube wall temperature (kg m^{-1} s^{-1})
Pr	Prandtl number evaluated at fluid bulk temperature (–)		

components and innovative design features that enhance the safety of the reactor [5]. Integral layout, pressurized water nuclear reactors are typically equipped with helically coiled steam generators, notably for their compactness, heat transfer efficiency and low susceptibility to thermal expansion concerns.

Due to its practical relevance, flow boiling in helically coiled tubes has been investigated quite extensively [6–12], showing that the heat transfer over most of the vapor quality range is due to both nucleate boiling and convection. Most authors found straight tube flow boiling correlations appropriate to predict the heat transfer coefficient during flow boiling in helically coiled tubes, although a general consensus on which straight tube correlations work best for helical coils has not been reached. Remarkably, some authors found straight tube flow boiling correlations not very accurate for helical coils [9,10], and proposed new correlations specific for flow boiling in helically coiled tubes. Notably, most of the available experimental studies have been carried out with comparatively short test sections, while the measurements taken with long, prototypical test sections that are required for the validation of system computer codes used in design, optimization and transient and safety analysis are still missing. To fill in this gap, the present study was conducted with a prototypical, full-scale 24 m long steam generator test tube, covering the operating conditions of medium pressure (2–6 MPa), low mass flux (200–800 kg m^{-2} s^{-1}) and low heat flux (40–230 kW m^{-2}) typical of the steam generators used in small modular nuclear reactor power plants. Moreover, this study includes an extensive assessment of 11 straight tube flow boiling correlations regarding their suitability for helically coiled tubes, using the large databank generated herein (1575 points). This study is part of a wide experimental program addressing single and two-phase boiling flows in helical coils, targeting specifically the design and operation of compact steam generators for nuclear power applications, notably small modular units [13–18].

2. Experiments

The test section used for the present experimental campaign was an helically coiled steam generator tube, made of stainless steel AISI 316 and framed into an open loop test facility built inside the boiler building of the "Emilia" power station in Piacenza-Italy (SIET Labs: www.siet.it), as shown schematically in Figs. 1 and 2. The test tube had internal and external diameters of 12.49 mm and 17.23 mm, respectively (corresponding nominal values were 12.53 mm and 17.15 mm), and a total length of 32 m. In particular, five tubes of 6 m each and one tube of 2 m were coiled and welded together to compose the test section tube. The variation of the tube external diameter along the test section due to the bending was found to be within 1–2%, small enough to neglect the tube ovalization in the data reduction and post processing. The average roughness of the tube inner surface was 3.1 μm (from six local measurements: maximum value 3.5 μm, minimum value 2.5 μm). The coil diameter and coil pitch were 1.0 m and 0.79 m, respectively, corresponding to 10 full coil turns and resulting in a total height of the steam generator tube of 8.0 m. The geometry of the test tube is fairly prototypical of steam generators for small modular pressurized water nuclear reactors of integral layout. The test section main geometrical data are summarized in Table 1.

As shown in Fig. 1, the test facility supply section fed the test section with demineralized, deionized water (mean electrical conductivity of 1.5 μS cm^{-1}) from a reservoir by means of a booster pump and a feed-water pump in series. The bypass line with a control valve located downstream of the pumps allowed the adjustment of the flow rate to the test section. A throttling valve was also included to avoid dynamic instabilities (density wave type) in the test section during operation. The subcooling of the water feeding the test section was adjusted with an electrical preheater located upstream of the test section (within 40–50 K in the present experiments). The test section was electrically heated via Joule effect with DC current. In particular, as shown in Fig. 2, three electrical power connections were provided: one at the inlet of the test section, one at the outlet of the test section and another one at an intermediate position along the test tube, located 24 m from the inlet and 8 m from the outlet of the test section. This allowed the heat flux in the last 8 m long portion of the test section to be varied independently from the heat flux delivered in the first 24 m long portion of the test tube, a feature that proved useful during post dryout studies (not discussed here) to better mimic the

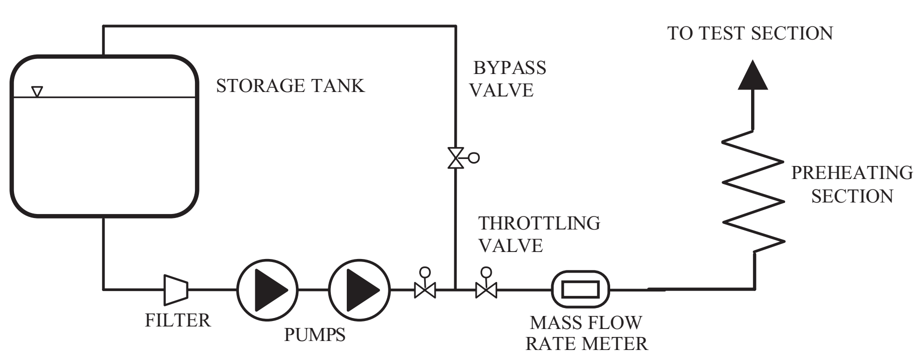

Fig. 1. Schematic representation of the test facility.