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Two-phase pressure drop prediction in helically coiled steam generators for nuclear power applications

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ABSTRACT

This study considers the prediction of the pressure gradient with water–steam two-phase flows through helically coiled steam generator tubes, focusing in particular on the operating conditions of low-medium pressure, low mass flux and low heat flux typical of once-through steam generators with in-tube boiling adopted in small modular nuclear reactor systems. Twenty-five widely used empirical correlations have been tested against an experimental pressure drop databank drawn together in this study containing 980 data points. Since no existing correlation is capable of collapsing and satisfactorily fitting the collected databank, a new pressure drop prediction method for helically coiled tubes is proposed. This new prediction method is very simple to implement, as it is based on the homogeneous flow model, is asymptotically consistent with straight tube two-phase flows and is largely superior in accuracy to existing prediction methods (mean absolute error of 7.3%, and 9 points out of 10 captured to within ±15%). The new prediction method is applicable for operating pressures in the range of 0.75–9.0 MPa, mass fluxes from 400 kg/m^2s to 1191 kg/m^2s, heat fluxes up to 750 kW/m^2, tube diameters within 5–20 mm and coil to tube diameter ratio above 32.4. Curvature effects on the pressure gradient in helical coil two-phase flows can be significant, particularly with high velocity flows in tight curvature coils where the centrifugal force is intense.

1. Introduction

Helically coiled tubes are frequently used in once-through steam generators with in-tube boiling due to the several advantages they offer in comparison with conventional straight tubes. For example, helically coiled tubes provide a greater heat transfer area per unit volume, which makes them attractive in applications where high compactness is desired, such as marine propulsion and integral layout nuclear reactors. Besides, helically coiled tubes can easily accommodate thermal expansions, which make them particularly robust in transient and off-normal operation. Moreover, the radial accelerations induced by the helical path promote liquid droplets de-entrainment from the vapor, thus keeping the tube internal surface wet to higher vapor qualities than in a straight pipe. As a consequence, the onset of dryout is delayed and the heat transfer effectiveness in the post dryout region is enhanced with respect to a straight pipe.

Building on the previous positive experience gained with advanced gas reactors and liquid metal fast reactors, helically coiled once-through steam generators with in-tube boiling are currently receiving a renewed interest, promoted by their frequent adoption in advanced small modular nuclear reactor systems (nuclear reactors with equivalent electric power output below 300 MW, according to the International Atomic Energy Agency [1]). Presently, there are about 50 different small modular nuclear reactor designs in various stages of development worldwide [2], including advanced water cooled reactors, liquid metal cooled reactors, high temperature gas cooled reactors, and molten salt reactors. The main advantages of small modular nuclear reactor systems with respect to conventional nuclear power stations include lower initial capital investment, enhanced passive safety, simplified design, proliferation resistance, greater flexibility of operation and capability to meet the smaller demand of power typical of emerging economies and developing countries [3,4]. Among the small modular nuclear reactor systems currently under development worldwide, the integral layout pressurized water reactors are the most mature designs and stand out for near- and mid-term commercial deployment. In integral layout pressurized water reactors, all the components usually associated with the primary

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Nomenclature

d	tube diameter (m)
D	coil diameter (m)
f_h	homogeneous Fanning friction factor (–)
f_{sf}	single-phase Fanning friction factor (–)
g	acceleration of gravity (m/s^2)
G	mass flux (kg/m^{2}s)
P	pressure (Pa)
Re_h	homogeneous Reynolds number (–)
x	vapor quality (–)
z	axial coordinate along channel (m)
μ_g	vapor viscosity (kg/ms)
μ_h	homogeneous viscosity (kg/ms)
μ_l	liquid viscosity (kg/ms)
ρ_g	vapor density (kg/m^3)
ρ_h	homogeneous density (kg/m^3)
ρ_l	liquid density (kg/m^3)
ϑ	channel inclination with respect to the horizontal (–)

circuit of a conventional pressurized water reactor are housed within a single reactor vessel, thus significantly reducing the likelihood of loss of coolant accidents and enhancing the primary circuit cooling capability in natural circulation. These designs combine well known technologies and several years of commercial operating experience accumulated with traditional pressurized water reactors, with novel components and innovative design features that significantly enhance the safety of the system [5]. Due to their compactness, heat transfer efficiency and low susceptibility to thermal expansions, helically coiled once-through steam generators with in-tube boiling are the preferred option for integral layout pressurized water nuclear reactors.

The accurate prediction of the pressure drop is crucial for a sound design and efficient operation of any two-phase flow systems, particularly so with nuclear power plant steam generators that are one of the most expensive components of the power plant, whose longevity and efficient operation are essential to profitable power production. In particular, the accurate prediction of the pressure gradient along the steam generator tubes is required to properly predict the temperature difference between the primary and the secondary fluid streams at the pinch point, which is one of the controlling parameters in the steam generator sizing. Notably, the emphasis on compactness that characterizes integral layout nuclear power plants makes the accurate prediction of the temperature difference at the pinch point crucial for a sound system design. Moreover, it is well known that once-through steam generators with in-tube boiling are vulnerable to undesired flow-pressure oscillations. The most efficient way to control these undesired oscillations is to apply orifices at the inlet of the steam generator tubes. The precise knowledge of the tube side pressure drop is a prerequisite for a sound design of the orifices to achieve stable flow conditions while avoiding overdesign that would result in a corresponding overdesign of the feed-water pump, thus eroding the system profitability.

Due to its practical relevance, two-phase flow pressure drop in helically coiled tubes has been investigated quite extensively [6–23]. Notably, several authors found straight tube prediction methods appropriate to predict the pressure drop in helical coils. In particular, the straight tube methods of Lockhart and Martinelli [24] and of Martinelli and Nelson [25] have been frequently reported to satisfactorily extrapolate to helically coiled tubes. Unfortunately, however, no systematic assessment of straight tube pressure drop correlations for use with helical coils has been conducted at present. Moreover, straight tube correlations have been invariably modified for use with helical coils by replacing the single-phase straight tube friction factor expressions of the original formulations with single-phase friction factor correlations specific for helical coils. This is normally a minor modification, as in turbulent flow conditions the difference in the friction factor between a straight tube and a helical coil is normally on the order of 10–20%. Nonetheless, this modification somewhat obscures the actual accuracy of existing straight tube pressure drop prediction methods when extrapolated to helical coils. What is worse, when modifying straight tube correlations for use with helical coils, different authors picked different single-phase friction factor correlations for helical coils, so that even comparing the results of different researches is rather difficult. Notably, some authors proposed pressure drop prediction methods specifically designed for helically coiled tubes [11,12,19,20,23,26]. None of these methods, however, is based on a large and diversified experimental databank, so that their applicability and accuracy in general design applications outside the operating conditions covered in the respective underlying databanks is unclear at the moment.

The cornerstone of the present study is the prediction of the two-phase pressure drop of water–steam flows in helically coiled tubes, focusing in particular on the operating conditions of low-medium pressure (1–8 MPa), low mass flux (200–1000 kg/m^{2}s) and low heat flux (100–500 kW/m^2) typical of the operation of the once-through steam generators with in-tube boiling used in small modular nuclear reactor systems. First, an experimental pressure drop databank put together with literature data is used to provide an extensive and critical assessment of existing pressure drop correlations: 19 widely used straight tube correlations and 6 more prediction methods specifically derived for helical coils. Then, a new pressure drop prediction method for water–steam flow in helically coiled tubes is proposed. This new prediction method is very simple to implement, as it is based on the homogeneous flow model, is asymptotically consistent with straight tube two-phase flows and is largely superior in accuracy to existing prediction methods. The present study is part of a wide research program addressing single and two-phase boiling flows in helically coiled tubes, focusing in particular on the design and operation of compact once-through steam generators with in-tube boiling for nuclear power applications, notably small modular units [22,23,27–31].

2. Experimental pressure drop databank

The main details of the experimental pressure drop databank collected from the open literature for use here are summarized in Table 1, while a selection of histograms that further describes the collected data is shown in Fig. 1. Even though the pressure drop in water–steam flows through helical coils has been studied quite extensively [6,8,11–13,19,20,22,23], only Zhao et al. [20] and Santini et al. [23] provide accurate pressure drop data in usable form. In particular, Zhao et al. [20] tested an electrically heated helical coil with a tube diameter of 9.0 mm and a coil diameter of 292 mm, corresponding to a coil to tube diameter ratio of $D/d = 32.4$. Their test section was 1.38 m long, was equipped with two pressure taps at the inlet and outlet to measure the pressure drop and was fed with two-phase flow generated in a preheater located upstream of the test section. On the other hand, Santini et al. [23] made experiments with an electrically heated helical coil with a tube diameter of 12.49 mm and a coil diameter of 1.0 m,