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Development and experimental validation of a computational model for a helically coiled steam generator

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A B S T R A C T

A computational model is developed to describe the thermo-fluid-dynamic behaviour of a helically coiled steam generator device working with water and widely adopted in the nuclear industry. The discretized governing equations are coupled using an implicit step by step method. The mathematical model includes: a subcooled liquid region, a two-phase flow region, and a superheated vapour region (according to the once-through nature of the heat exchanger). All the flow variables (enthalpies, temperatures, pressures, vapour qualities, velocities, heat fluxes, etc.), together with the thermo-physical properties, are evaluated at each point of the grid in which the domain is discretized. A full-scale experimental investigation carried out at SIET thermal-hydraulics labs in Piacenza (Italy), and aimed at characterizing the fluid-dynamic behaviour of two-phase flows in helically coiled tubes, is referenced in the present paper. Two-phase pressure drops data reduction allowed optimizing a suitable form of the friction factor multiplier required by momentum balance equation. Comparisons of the numerical simulations with a wide range of two-phase pressure drops measurements (experiments conducted both in diabatic and adiabatic conditions) are shown in order to validate the proposed model.

1. Introduction

Helically coiled once-through boilers have been widely used in the past for several power conversion systems, ranging from fossil fuelled power plants, natural gas liquefaction apparatus, to solar energy concentrator receivers, and steam generators for nuclear reactor applications as well. The main advantages of such tubular configuration derive from a higher compactness, adaptability to cylindrical shape and an excellent behaviour in presence of thermal expansions (accommodated, as a matter of fact, behaving as a spring). High efficiency heat transfer is obtained by combining the positive features of a local cross-flow disposition with a global counter-flow along the exchanger tubes [1,2]. Different nuclear power plants adopted in the past a helical-coiled steam generator to drive the turbine: gas-cooled reactors (AGR, Fort St. Vrain HTGR, THTR-300), the SuperPhoenix fast reactor (sodium cooled), as well as one water cooled nuclear reactor for ship power propulsion (Otto Hahn). Nowadays, helically coiled steam generators are envisaged within the innovative projects to provide a more safe and sustainable nuclear energy option, both with near-medium term deployment (Generation III + systems [3]) and long term deployment (Generation IV systems [4]).

The hydrodynamics in a coiled tube is complicated with respect to straight tubes due to the presence of secondary flows. Moreover, the introduction of two new geometrical parameters, namely the coil diameter and the coil pitch, renders much more difficult the prediction of most important thermal-hydraulic variables, such as heat transfer coefficients and pressure drops (friction term), in particular in the two-phase region. In other respects, a reasonably accurate prediction of the pressure drops in a steam generator, for instance, is fundamental for calculating the total head of the feedwater pump of the Rankine cycle, as well as for calculating the thermodynamic conditions of the steam at turbine inlet and optimizing thus its efficiency. Detailed knowledge of the various thermal-hydraulics phenomena in a steam generator is necessary in order to fulfill a complete design of the component. To this aim, more general and accurate methods for predicting thermal and fluid-dynamic behaviour of a two-phase flow within a helical-coiled tube are required [2,5].

Numerous experimental studies on helically coiled systems have been reported in the open literature. A pioneering work in this field is due to Ruffel [6], who tested three coil curvatures with electrically heated test sections for AGR performances prediction.

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