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Research Paper

Helical coil thermal-hydraulic model for supercritical lead cooled fast reactor steam generators

Marco Caramello *, Cristina Bertani, Mario De Salve, Bruno Panella

Polytechnic of Turin, Corso Duca degli Abruzzi, 24, 10129 Turin, Italy

HIGHLIGHTS

- A model for thermal-hydraulic analysis of helical coil steam generators is developed.
- Liquid metal and supercritical water can be simulated.
- Preliminary qualification on nominal data of BREST reactor is satisfactory.

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ABSTRACT

This paper deals with the presentation of the developments of an in-house code created for the steady state thermal-hydraulic analysis of helical coil steam generators. These components are of particular interest in the nuclear field both for normal and emergency operation because they are able to remove thermal power with a very high degree of compactness. This quality is particularly attractive for small modular reactors where constraints related to the volume of components are a critical issue. The model is able to perform calculations to validate the design of these components, run fast sensitivity analyses and show the spatial behaviour of the most important parameters of the flow such as temperature, pressure, heat transfer coefficients and heat flux. In this paper the additional features of the model such as the possibility to simulate supercritical water and liquid metals are presented. The choices made for the selection of semi-empirical correlations available in literature for the estimation of the heat transfer coefficients in these conditions are described and justified. Finally, the predictive capability of the updated model is validated against the nominal data of the BREST reactor, which is a lead fast reactor that make use of supercritical water.

1. Introduction

Helical coil steam generators (SG) have been successfully used in nuclear industry in the past in some reactors, in particular, for the first fast reactors, like Monju or Superphoenix [1]. In recent years, helical coil steam generators were chosen for some lead fast reactors like the Russian project Brest [2] and in some small modular reactors (SMR) like Carem, Smart, Iris and NuScale [3–6]. These components are the heat sink of the thermal power of nuclear core during normal operation. In case of loop-type reactors the shell side of the SG is connected to the main vessel by means of piping whereas in pool-type or SMR reactors the bundle is housed inside the main vessel with all the components of the primary system.

These components are an attractive solution to remove higher values of thermal power in relatively small volumes, thanks to the high degree of compactness. The geometry guarantees higher heat transfer surfaces per unit length compared with standard straight tube heat exchangers. The centrifugal forces acting on the fluid inside the helical ducts tend to laminarize the flow, enhance the heat transfer coefficient and increase the frictional pressure drops.

Among the activities in support of the development of such components there is the implementation of codes and models for the design and verification of mechanical and thermal-hydraulic features. Thermal-hydraulic models have the aim to support the SG sizing and design verification for nominal, abnormal, stationary or transient conditions. Some works are available on literature on the development and subsequent validation of in-house models and of codes for helical coil steam generators. Sakai et al. [7] studied the temperature profiles of helical coil steam generators. They developed a multi-shell model dividing the component in different calculation regions on the basis of the coils position. They validated the model by comparing their results with experimental data of a 50 MWth steam generator that employed water and sodium as fluids. The results showed that a multi-shell method can increase the accuracy in the calculation of temperature profiles. Mochizuki [8] validated the NETFLOW code by comparing the results

* Corresponding author. Tel.: +0110904408.
E-mail address: marco.caramello@polito.it (M. Caramello).

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with respect to experimental data of a 50 MWth steam generator and MONJU fast reactor turbine test trip. The results showed the general capability of the code to predict the performance of the component both in stationary and transient condition, and thanks to the calculation speed of the code, the author suggests to use the model also for education purposes. Caramello et al. [9] developed a model able to study the steady state operation of an helical coil steam generator. The predictive capabilities of the model have been tested with reference to the nominal data of IRIS steam generator and the results obtained with the system code RELAP5/Mod.3.3. The results showed a good agreement with respect to the nominal data and the results from the system code apart from the dryout region. Considering the importance of helical coil heat exchangers for small modular reactors many codes are equipped with dedicated thermal-hydraulic modules for these components: two of them are TAPINS and TASS/SMR [10–13]. TAPINS is a monodimensional system code developed at Seoul National University based on one dimensional momentum integral model that has been validated against experimental data from MIT and REX-10 facility. TASS/SMR is an extensively validated one dimensional system code developed at KAERI that makes use of the conservation equations of mass, momentum, energy and noncondensables and is used for the design and analysis of the SMART reactor.

The aim of the work is to present a new thermal-hydraulic model for the steady state characterization of helical coil heat exchangers. The model can simulate different fluids such as water (subcritical and supercritical) lead, LBE and sodium and lends itself well to study the heat removal of these components in SMR and innovative fast reactors. The model iteratively solves the steady state solution of the thermal-hydraulic quantities of the flow such as temperature, pressure, heat flux, heat transfer coefficients, flow quality and void fraction showing their behaviour along the component. Specific semi-empirical correlations available in open literature serve to estimate convective heat transfer coefficients and friction factors. The model is particularly suitable to run fast sensitivity analyses on the geometry and the conditions of the fluids. It has also a library with the behaviour of thermal conductivity with temperature for different materials. In the following paper the mathematical model implemented is described and particular emphasis is given to the most recent updates (i.e. the capability to simulate supercritical water and liquid metals). Finally, a preliminary qualification of the model is carried out by comparing its results against the nominal data of the helical coil steam generator of the BREST reactor, which is a lead fast reactor that makes use of supercritical water as secondary fluid.

2. Model description

The control volume devoted to a helical coil steam generator is generally cylindrical, being it inside the reactor pressure vessel in case of SMR reactors or external to it in case of large scale reactors. The single helixes through which the water flows are set in parallel inside of it alternating clockwise and anticlockwise rotation with respect to the axis of symmetry of the cylinder to maximize the compactness. Helical pipes are set on several rows, and the pipes belonging to the same row have the same diameter. The rows are limited by a metallic sheet at the boundary which constitutes the end of the cylinder and permits to direct the external fluid which flows in the interstices between the helical pipes. An example of the geometry described considering a single row is depicted in the cutaway of Fig. 1.

The prevalent direction of the fluid inside the coils is ascending. As it moves, it is heated by the external fluid: four characteristic regions are encountered in the case of subcritical fluids, namely the subcooled region, the subcooled boiling region, the saturated region and the superheated one. The motion of the external fluid is from the top to the bottom of the steam generator. As the fluids descend in the steam generator the flow is disturbed by the presence of the coils causing a change in the motion and enhancing the transversal mixing with respect to the prevalent motion direction. The definition of the geometry inside the model and its subsequent discretization is made on the basis of several simplifications with respect to the real configuration. The methodology through which the geometry is defined has the aim to create a monodimensional model of the heat exchanger sufficiently accurate to describe the general performance of the component. The assumptions on which the geometry is created are listed below and are generally applicable for the steam generators used in the nuclear field:

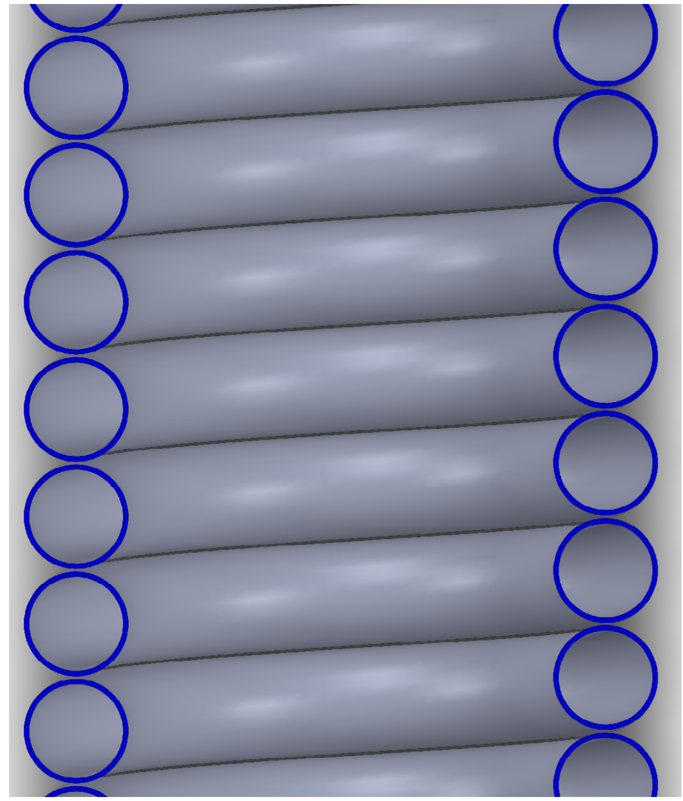

Fig. 1. Cutaway of a helical coil row.

- the helical bundle is approximated with a single pipe whose length is equal to the average length of the coils,
- the external fluid is simulated as a vertical annular channel which transfers heat with the pipe, which is equivalent to consider a perfect mixing on the horizontal cross section.

By simulating the component in this way the information related to local effects like inlet/outlet section or the different weight of centrifugal forces in the bundle is lost; however, such effects can be neglected if the dimension of the fully developed region is the predominant part of the steam generator and some relevant parameters for the helical coils are nearly constant, like the curvature ratio (δ) or the torsion ratio (τ), whose definition is recalled in equations 1 and 2. The approximation of single averaged values for curvature and torsion ratios still permits to consider in the calculation the macroscopic effect of the centrifugal forces both on the heat transfer and pressure drops of the fluid inside the helical coil and the pressure loss of the external fluid. These are the most impacting

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