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1/3D modeling of the core coolant circuit of a PHWR nuclear power plant

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ABSTRACT

A multi-dimensional computational fluid dynamics (CFD) one-phase model to simulate the in-core coolant circuit of a pressurized heavy water reactor (PHWR) of a nuclear power plant (NPP) was performed. Three-dimensional (3D) detailed modeling of the upper and lower plenums, the downcomer and the hot and cold leg nozzles was combined with finite volume one-dimensional (1D) code for modeling the behavior of all the 451 coolant channels. Suitable functions for introducing the distributed (friction losses) and concentrated (spacer grids, inlet restrictors and outlet throttles) pressure losses were used to consider the local pressure variation along the coolant channels. The special power distribution at each coolant channel was also taken into account. Results were compared with those previously obtained with a 0/3D model getting more realistic temperature patterns at the upper plenum. Although the present model is restricted to one-phase phenomena, the prediction of the local pressure and temperature along the channels allows for a preliminary identification of the location of incipient boiling by comparing with the local saturation temperature. The present model represents an improvement with respect to the previous 0/3D model. It corresponds to the necessary step before achieving a 1/3D two-phase model with which the pressure drop and subcooled boiling along the coolant channels as well as the overall reactor pressure vessel (RPV) void fraction distribution can be evaluated more accurately.

1. Introduction

The currently being built nuclear power plant (NPP) Atucha II (CNA II), is a pressurized heavy water reactor (PHWR) with a projected total thermal power of 2160 MWt and electric power of 745 MWe. In contrast to the pressure tube type CANDU reactors, CNA II is a pressure vessel type. The core has a vertical configuration housing 451 cooling channels (CC) housed in the moderator tank. Each CC has the aim to remove the thermal power generated by fission of atoms through a coolant flow pumped under high pressure from the lower-plenum to the upper-plenum. The fuel bundles are composed by a set of 37 fuel rods of 5.3 m active length with 13 spacer grids to strengthen and lining up the fuel assembly. The CCs are arranged in a 272 mm trigonal lattice pitch within the moderator tank. CNA II will employ a fuel composed of natural uranium and deuterium (D_2O) heavy water (HW) for cooling and moderation purposes. During normal operation the fuel elements follow a replacement strategy, in which their position is periodically changed with the aim to control burnup.

The NPP Atucha I (CNA I) and CNA II are one of a kind NPP around the world. CNA I is in operation since 1974 and therefore, valuable experience has been gained. However, CNA II is two times larger than the first one, resulting in a technological challenge. For this reason, the main objective of this study is to provide useful information about the thermo-hydraulic behavior in the reactor pressure vessel (RPV) of CNA II.

A 3D drawing of the coolant circuit of CNA II is shown in Fig. 1. In each one of the two loops the hot coolant flows through the hot leg from the RPV to the steam generator to transfer the heat to the secondary circuit. After that, the cold coolant flows to the pump to recover the high pressure and then it returns to the RPV through the cold leg.

The two pair of hot and cold legs are placed diametrically opposed in the RPV. The secondary circuit of light water (LW) is preheated by extracting heat from the moderator circuit and then its temperature continues to rise in the steam generators so as to obtain steam in saturated conditions.

The coolant circuit inside the RPV can be divided into two main reservoirs, the downcomer and lower plenum and the upper

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Glossary

NPP	nuclear power plant
CNA	nuclear power plant Atucha
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor
0D	zero-dimensional
1D	one-dimensional
3D	three-dimensional
CC	coolant channel
HZ	hydraulic zone
RPV	reactor pressure vessel
MFR	mass flow rate
NPP	nuclear power plant
CFD	computational fluid dynamics
LOCA	loss of coolant accident
LW	light water
HW	heavy water
SSP	sink/source point
PISO	pressure implicit with split operator
SIMPLE	Semi-Implicit Method for Pressure-Linked Equations
PIMPLE	hybrid PISO/SIMPLE algorithm
VFM	volume finite method
UFR	User Fortran Routine
x	axial position
U	velocity
p	pressure
T	temperature
T_{sat}	saturation temperature
ρ	density
g	gravity acceleration
h	enthalpy
$\bar{\bar{\tau}}$	stress tensor
μ	dynamic viscosity
κ	thermal conductivity
S_S	mass sink/source
S_M	momentum sink/source
S_E	energy sink/source
k	turbulent kinetic energy
ε	turbulence dissipation rate
μ_t	turbulent viscosity
G_t	turbulence production
η	wall roughness
C_f	Darcy friction factor
C_{sp}	pressure drop coefficient at spacer grid
C_{ir}	pressure drop coefficient at inlet restrictor
C_{ot}	pressure drop coefficient at outlet throttle
S_f	pressure drop by wall friction
S_{sp}	pressure drop by the spacer grids of the coolant channel
S_{ot}	pressure drop by the outlet throttle of the coolant channel
S_{ir}	pressure drop by the inlet restrictor of the coolant channel
S_{ps}	power source
D_h	hydraulic diameter
Re	Reynolds number
V_p	cell volume
A_f	cell flow area

Fig. 1. CNA II main coolant circuit of the PHWR.

plenum, both connected through the CCs and the bypass tubes which transport less than 3% of the total flow. The coolant enters to the RPV through two cold legs and travels down towards to the lower plenum through the annular downcomer between the pressure vessel and the moderator tank walls. Fig. 2 shows a cross section view combining two cutting planes to visualize one hot leg and one cold leg of the RPV (left).

The coolant enters to the CCs through inlet nozzles placed at the lower plenum. The 451 CCs are grouped in 5 hydraulic zones (HZ) with different mass flow rate (MFR) attending to the radial power distribution of the reactor. This in-core flow distribution is mainly produced by flow restrictions placed at some of the CC inlets. Then, the coolant flow is directed upwards through the CCs where the fission heat is extracted and later on it flows towards the upper plenum through vertical slots situated at the CC tubes (outlet throttles). The pressure drop along the CCs is mainly caused by concentrated form losses (sudden area change) at the inlet nozzles, the outlet throttles and the spacer grids. Therefore, friction losses along the fuel rod and tube channel walls are also significant for the HZ with the highest MFR. The expected pressure drop for the coolant flow along the CCs shall be around 6 bar, while the total pressure drop along the whole RPV is around to be around 7.3 bar.

The lower plenum has a flow distributor composed of rhomboidal cells housing the CC inlets. Each cell can group up to 9 CCs (see Fig. 2right). The upper plenum has a convex ellipsoidal shape housing 9 hafnium and 9 steel control rods. Moreover, the upper plenum is crossed by the 451 CC tubes and the moderator inlet and outlet ducts. All these components affect the flow and the thermal distribution in the upper plenum.

In Fig. 2, the scratched solids above the upper and below the lower plenums are the filling bodies, which serve to reduce the coolant inventory in the reactor coolant circuit.

Fig. 3 shows the HZ distribution. The HZ 5 is the most important one, containing 253 of the 451 CCs with around 70% of the total coolant flow. The main aim of the in-core flow distribution is to obtain a balance between the fission heat released and the MFR in each CC.

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