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Stability analysis of the Supercritical Water Reactor by means of the root locus criterion

E. Cervi, A. Cammi*

Politecnico di Milano, Department of Energy, Nuclear Engineering Division, via La Masa 34, 20156 Milano, Italy

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ABSTRACT

The Supercritical Water Reactor (SCWR) is a concept for an advanced nuclear reactor operating at high temperature (500 °C average core outlet temperature at nominal power) and at high pressure (25 MPa), which give the SCWR a thermal efficiency of about 45%. However, due to the strong variability of the water properties near the thermodynamic pseudocritical point, concerns are raised towards thermal-hydraulic instabilities. A simulation tool was developed in Matlab® from the perspective of linear systems, aimed at investigating the reactor stability and identifying potential regions of instability through a consolidated and relatively simple approach. A frequency-domain stability analysis of the SCWR is carried out with the root locus criterion, characterizing the system stability features over its entire operating power interval. The impact of the coolant flow rate on the stability is also studied. The results show that the system is stable over the whole investigated operational range. Finally, the dynamic behavior of the SCWR is compared to the Boiling Water Reactor (BWR), pointing out significant differences due to the different working points and design features of the two reactors. The results of this study could be a starting point for further research on the SCWR, providing the designers with important feedbacks for the optimization of the SCWR coolant circuit.

1. Introduction

The Supercritical Water Reactor (SCWR) is one of the six concepts under investigation in the GEN-IV international advanced reactor development program. It is a combination between the traditional Light Water Reactor (LWR) and the supercritical Fossil Power Plant (FPP).

Water at a pressure and at a temperature above its critical point (p_c = 22.06 MPa, T_c = 373.9 °C) is called supercritical. The technology of supercritical water used as coolant is well established in the field of Fossil Power Plants, allowing to reach a larger thermal efficiency due to the increased pressure and temperature of the fluid.

Since no boiling takes place in the SCWR, the reactor can be operated at high temperature without any concern about the CHF (critical heat flux), which limits the operating temperature of the operating temperature of the traditional LWRs. For this reason, the system reaches a thermal efficiency of about 45%, significantly higher than the current LWRs and comparable to those of modern Fossil Power Plants (Ortega Gómez, 2009). Moreover, compared to a Boiling Water Reactor, steam separators and recirculation pumps are no longer needed, allowing significant plant simplifications and a more compact design.

One of the major concerns about the SCWR is represented by thermal-hydraulic instabilities, due to the strong variability of the water density near the thermodynamic pseudocritical point. In fact, owing to the large temperature difference between the core inlet and outlet, the water density decrease in the SCWR core is even larger than in BWRs. Hence, even if no phase transition takes place, all the instability phenomena occurring in two-phase flows can also be observed in the SCWR.

So far, the problem of thermal-hydraulic instabilities in supercritical water reactors has been addressed by many authors. Chatoorgoon (2001) studied the stability of a supercritical fluid flow in a natural circulation CANDU-X supercritical reactor (Dimmick et al., 1998) using a non-linear numerical code. Yi et al. (2004) developed a linear stability analysis code in frequency domain to study the thermal-hydraulic stability of the SCWR-H, a thermal-spectrum supercritical reactor, by means of the decay ratio. Cheng and Yang (2008) developed a point-hydraulics model to study the onset of self-sustaining flow oscillations in a supercritical cooling loop, based on the definition of suited dimensionless numbers. Ortega Gómez et al. (2008) studied the linear stability characteristics of a uniformly heated supercritical channel by evaluating the eigenvalues of a one-dimensional model. A time-domain analysis of non-linear phenomena was also presented. Moreover, a stability analysis of the SCWR, based on the U.S. reference design, was carried out by Zhao (2005). Further investigations on core-wide

* Corresponding author.
E-mail address: antonio.cammi@polimi.it (A. Cammi).

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Nomenclature

Latin symbols

A_c	total coolant passage surface, m^2
A_w	total water rod passage surface, m^2
c	specific heat, $J\,kg^{-1}\,K^{-1}$
D_h	hydraulic diameter, m
F	Darcy–Weisbach factor
G	channel inlet mass flow rate (Appendix A), $kg\,s^{-1}$
g	gravitational acceleration, $m\,s^{-2}$
$\hat{h}$	specific enthalpy, $J\,kg^{-1}$
h_c	convective heat transfer coefficient between the coolant and the water rod wall, $W\,m^{-2}\,K^{-1}$
h_f	convective heat transfer coefficient between the fuel and the coolant, $W\,m^{-2}\,K^{-1}$
h_w	convective heat transfer coefficient between the water rod wall and the water rod, $W\,m^{-2}\,K^{-1}$
K	concentrated pressure drop coefficient
k	thermal conductivity, $W\,m^{-1}\,K^{-1}$
m	mass, kg
$\dot{m}$	mass flow rate, $kg\,s^{-1}$
Δp	pressure drop, Pa
Pr	Prandtl number
$\dot{Q}$	thermal power, W
q'''	volumetric power source, $W\,m^{-3}$
Re	Reynolds number
R_{pin}	fuel pin radius, m
S_c	total external water rod surface area, m^2
S_f	total fuel surface, m^2
S_w	total internal water rod surface area, m^2
T	temperature, °C
t	time, s
V	volume, m^3
Δz	height of a channel sub-volume, m

Special symbols

α_c	coolant density reactivity feedback coefficient, $pcm\,m^3\,kg^{-1}$
α_f	fuel Doppler reactivity feedback coefficient, $pcm\,K^{-1}$
α_w	water rod density reactivity feedback coefficient, $pcm\,m^3\,kg^{-1}$
β	total delayed neutron fraction, pcm
β_i	i-th precursor group delayed neutron fraction, pcm
η_i	i-th group normalized precursors density
Λ	invariant neutron average lifetime, s
λ_i	i-th precursor group decay constant, s^{-1}
μ	dynamic viscosity, Pa s
ξ	derivative of the density with respect to the temperature, $kg\,m^{-3}\,K^{-1}$
ρ	density, $kg\,m^{-3}$
ρ	reactivity, pcm
ψ	normalized neutron density

Subscripts-superscripts

b	bulk-fluid
c	coolant
DC	downcomer
ev	exit valve
f	fuel
i	i-th sub-volume
in	inlet
iso	isothermal
LP	lower plenum
o	steady state
out	outlet
p	constant pressure
pc	pseudo-critical
$pump$	pump
s	surface
SL	steam line
w	water rod

instabilities in supercritical water reactors were proposed by Zhao et al. (2008a,b). Moreover, Zhao et al. (2011) presented a stability analysis of the U.S. SCWR based on a response matrix approach (see Fig. 1).

Cai et al. (2009) studied the stability of a supercritical water fast reactor (SWFR) during the power-raising phase at startup using the decay ratio. Liu et al. (2014) and Hou et al. (2011) developed a frequency-domain model for linear stability analysis, replacing the decay ratio (DR) with the logarithmic decay ratio (LDR) and also presented a time-domain model for nonlinear analysis. T'Joen and Rohde (2012) performed an experimental study of coupled neutronic and thermal-hydraulic stability of a natural circulation HPLWR, a European concept design for a SCWR. A scaled model of the HPLWR was used, with Freon R23 as a scaling fluid. Xiong et al. (2012) presented another experimental study of thermal-hydraulic instabilities in two parallel channels with supercritical water. Li et al. (2014) modeled thermal-hydraulic instabilities in supercritical reactors using APROS, a simulation software for full-scale modeling and dynamic simulation of industrial processes. Further experimental and numerical 3-D investigations of flow instabilities in two parallel channels with supercritical water were performed by Xi et al. (2014a,b). Dutta et al. (2015a,b) developed a one-dimensional thermal-hydraulic model to study single channel and parallel channels flow instabilities in the CANDU supercritical reactor, without considering the neutronic coupling. Finally, Shahzad et al. (2016) studied the single channel thermal-hydraulic stability of the CSR1000, a Chinese supercritical reactor, using a system response matrix method and evaluating the decay ratio.

The distinguishing feature of the current research, with respect to the other works available in literature, is that the SCWR stability is studied with the root locus criterion. With this method, the stability of a nuclear reactor can be characterized over its entire operational power range. In addition, root loci can be obtained for any other operational and geometrical parameter, as described in Appendix A. Hence, the proposed method constitutes a flexible and straightforward approach to study the plant dynamic behavior in different operating conditions, allowing to compare different design alternatives and possibly resulting in an important feedback for the designers. Finally, the dynamic behavior of the SCWR is also compared to a typical BWR, in order to point out eventual differences due to the different properties of boiling and supercritical water as well as to the different design features of the two systems.

2. Thermal-hydraulic instabilities

Many types of thermal-hydraulics instability phenomena are known in supercritical fluid flows. Among them, Density Wave Oscillations (DWOs) are one of the most concerning, since they may lead to unwanted mechanical vibrations or prejudice the heat transfer characteristics of the systems. DWOs consist in an oscillatory response of the