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The stability-analysis code FIAT development for density wave oscillations and its application to PV/PT SCWR

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ABSTRACT

SuperCritical Water Reactor owns a high operation pressure and a wide coolant temperature range. The thermo-physical properties and transport properties of the coolant in the core would change greatly both in the operation condition and startup process, which may causes density wave oscillations. In this paper, a stability analysis code named FIAT is developed based on frequency domain method to analyze the density wave oscillations (DWO) for SCWR. Frequency domain method and assumptions of constant pressure drop between core inlet and outlet are applied in the code. The neutron/thermal-hydraulic coupling characteristic equation is derived through using linearization and laplace transformation for the mass, momentum and energy conservation equations. With analyzing the characteristic equation, the onset of DWO instabilities can be confirmed with non-dimensional parameters plane. The prediction accuracy of the FIAT code is verified through experiment data and code-to-code comparison under subcritical and supercritical pressure. The FIAT code is applied to evaluate the effects of basic core model on the stability for pressure vessel type SCWR and optimize the startup procedure for pressure tube type SCWR. The results show that for pressure vessel type SCWR, fuel rods heat transfer model and neutron kinetics model help stabilize the system while water rods model produces an opposite effect. For pressure tube type SCWR, two pressurization phases are required during sliding pressure start-up process to meet both the thermal–hydraulic and stability requirements.

1. Introduction

The supercritical water cooled reactor (SCWR), which is the only water cooled reactor in the Generation IV reactor concepts, has been conceptually studied by many countries for its advantages on system simplicity and high efficiency (Heusener et al., 2000). There are mainly two types of SCWR concepts (Duffey et al., 2005): (a) a large reactor pressure vessel containing the reactor core (fueled), analogous to conventional PWRs and BWRs, such as SCLWR (Oka et al., 2002) and SCPR (Kataoka et al., 2002) in Japan; SCWR (Buongiorno and MacDonald, 2003) in U.S.; HPLWR (Starflinger et al., 2003) in Europe; CSR1000 (Liu et al., 2015; Wu et al., 2014) in China. and (b) distributed pressure tubes or channels containing fuel bundles, analogous to conventional CANDU and RBMK nuclear reactors, such as Canadian SCWR in Canada (Leung, 2011).

The low coolant flowrate and large density change in SCWR may cause undesirable flow instabilities in the fuel channels. Flow instabilities in reactor can disturb the control and the operation of the reactor. Thus careful design and analysis must be done to ensure that these instabilities can be detected. The instabilities in SCWR can be subject to the same types of instabilities in BWR because the dynamics of SCWR is similar to that of BWR. This study will consider only density wave instability (DWO), which is the most important instability type in reactor design.

Substantial works have been done on the linear stability analysis for density wave oscillations of BWR and plenty of literatures can be found (Kao, 1996; Hanggi, 2001). Compared to the long history of subcritical pressure stability study, DWO study under supercritical pressure has begun in the last decade. Yang (2003) analyzed the instability of vertical flow heating pipe under supercritical pressure, and adopted frequency domain linear analysis method to get the nuclear/hydraulic coupling instability. Tin et al., 2005 carried out linear analysis on the flow stability for Super LWR to investigate the thermal-hydraulic phenomena in an upward flowing heated channel as well as coupled nuclear/thermal-hydraulic in stabilities. The effects of water rods are also taken into account. Zhao et al., 2005 has done extensive researches on the stability analysis of US SCWR based on three-region model.

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Nomenclature

Symbol

A	Pseudo-saturated liquid point of three-region model in supercritical pressure
A_c	Flow area (m^2)
C_i	Concentration of ith delayed neutron precursor ($n\,m^{-3}$)
C_p	Specific heat capacity at constant pressure ($kJ\,kg^{-1}\,K^{-1}$)
De	Hydraulic equivalent diameter (m)
f	Friction coefficient
G	Mass flux ($kg\,m^{-2}\,s^{-1}$)
g	Gravitational acceleration ($m\,s^{-2}$)
h	Enthalpy ($kJ\,kg^{-1}$)
$\bar{h}$	Heat transfer coefficient ($W\,m^{-2}\,K^{-1}$)
k	Thermal conductivity of solid ($W\,m^{-1}\,K^{-1}$)
L	Heating length (m)
$Nexp$	Expansion Number
$Npch$	Phase Change Number
$Npsub$	Pseudo-subcooling Number
$Nsub$	Subcooling Number
n	Neutron density ($n\,m^{-3}$)
Ph	Wetted perimeter (m)
p	Pressure (MPa)
q''	Surface heat flux ($W\,m^{-2}$)
R	Ideal gas constant ($J\,mol^{-1}\,K^{-1}$)
r	Radius of surface (m)
T	Temperature (K)
t	Time
u	Velocity ($m\,s^{-1}$)
v	Specific volume ($m^3\,kg^{-1}$)
α	Vapor void fraction
Δx	Axial space increment
Λ	Prompt neutron lifetime (s)
β_i	Effective delayed-neutron fraction for the i th group
λ_i	Decay constant for i th delayed-neutron group (s^{-1}), length (m)
ρ	Density($kg\,m^{-3}$), Reactivity
Γ	DWO transfer function

Subscripts

1,2,3	Fuel pellet, inner cladding, outer cladding
A	Pseudo-saturated liquid in subcritical pressure
B	Pseudo-saturated steam in subcritical pressure
C	Clad
cool	Coolant
F	Fuel
f	Saturated liquid in supercritical pressure
g	Saturated steam in supercritical pressure
in	Inlet fluid
low	Lower plenum
m	Moderator
out	Out fluid
or	Inlet orifice
w	Wall
wa	Water rod

Acronyms

ATHAS	Advanced Thermal-Hydraulics Analysis Subchannel
BWR	Boiling Water Reactor
DWO	Density-Wave Oscillations
HEM	Homogenous Equilibrium Model
SCWR	Supercritical Water Cooled Reactor
PT SCWR	Pressure Tube-type Supercritical Water Cooled Reactor
PV SCWR	Pressure Vessel-type Supercritical Water Cooled Reactor
RBMK	graphite-moderated nuclear power reactor designed and built by the Soviet Union

Comparisons between SCWR and typical BWR stability characteristic and a stable start-up procedure for US SCWR are achieved in his doctor thesis. Gómez et al., 2008 analyzed DWO of supercritical water under the uniform heating condition. The results of steady state, linear and non-linear stability analysis are achieved. Chatoorgoon, 2008 analyzed the instability of two level parallel channel. The analysis result showed that the supercritical flow instability is mainly affected by the influence of the variation trend of density based on the changing enthalpy. Ambrosini, 2009 adopted the dimensionless parameter method to analyze the instability of heating supercritical fluid and proposed a new dimensionless parameter based on the pseudo critical temperature point property. This paper aims to develop a stability analysis code based on frequency domain method to study the density wave oscillation instability of SCWR. The code applies the homogenous thermal-hydraulic models which is coupled with fuel conduction model, neutron kinetics model, and three-region model water property to derive the characteristic equations for SCWR system. The susceptibility of stability to each code model such as fuel conduction model, point neutron kinetics model and water rod model is performed and advice for improving stability is proposed for CSR1000. The FIAT code is also applied to optimize the start-up procedure for Canadian SCWR.

2. Development of stability-analysis code FIAT

During startup and normal operation of SCWR, the coolant density will experience significant change while the temperature crosses the pseudo-critical point or saturated line, which may cause the density wave oscillations inside the core. A stability-analysis code FIAT based on frequency domain method is developed to investigate the SCWR density instability in this chapter. The basic models of the core should be processed through linearization and laplace transformation to get the relationships of pressure drop perturbation and inlet coolant velocity perturbation, of which the eigenvalues will be analyzed to determine whether the system is stable or not.

2.1. Basic thermal hydraulic model

Fig. 1 shows the schematic diagram of coolant flow inside CSR1000 (Wu et al., 2014). The core is made up of fuel bundle, coolant channel, water rod, heat structure between coolant channel and water rod, downcomer, lower plenum. The coolant entering the core will be separated into two parts, one part flowing down through the water rod and mixing with the other part flowing through the downcomer in the lower plenum. Then the coolant in the lower plenum flows through the coolant channels and remove the heat from the fuel bundles. The flow distribution inside Canadian SCWR is almost the same as that of CSR1000, except that downcomer doesn't exist in Canadian SCWR. All the inlet coolant of the Canadian SCWR enter the central tube of the pressure tube, which is similar to the water rod in CSR1000, reverses the flow direction at the bottom of the central tube and flows upward through the coolant channel (Domínguez et al., 2016).

The coupling relationship of the neutron/thermal hydraulic model is shown in Fig. 2. The neutron kinetics model will provide the value of core power q_f'' to the fuel conduction model, while