SYNOPSIS

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Thesis Title: **Analysis and Scaling of Coupled Neutronic Thermal Hydraulic Instabilities of Supercritical Water-cooled Reactor**

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Present thesis work primarily focuses on the analysis of flow instabilities in one of the most powerful concepts under Gen.-IV nuclear reactor technology, namely the supercritical water-cooled reactor (SCWR). Safety is the primary concern in nuclear reactors and the study of flow instabilities is an important aspect of safety assessment. To facilitate the study of complex phenomena in the laboratory, a method for designing a downscaled model has been proposed here for both natural as well as forced circulation SCWR. For detailed analysis of the stability characteristics of the system, a simple but computationally inexpensive model has been developed as lumped parameter model (LPM). Using this model, linear and nonlinear stability analyses have been done for various ranges of parameter values. Instabilities in SCWR subjected to seismic effects have also been analysed by using the LPM.

Nuclear power plants use the heat generated from nuclear fission in a contained environment to convert water to steam, which powers generators to produce electricity. Light water reactors (LWR) use water as coolant and moderator. SCWR is a Gen.-IV LWR. It is a concept for an advanced reactor that operates at supercritical pressures and temperatures (25–30 MPa, 500–520 ^{o}C exit temperature). Such a high coolant temperature at turbine inlet provides high thermal efficiency (~42%) which is substantially greater than any other LWR (~30%). Consequently, it also experiences significant density difference throughout the coolant channel, from the inlet to the outlet (720 kg/m^3 to 90 kg/m^3), which raises grave concerns about flow instabilities in the SCWR. To ensure a proper design of SCWR without any safety issues, detailed stability analysis over wide parameter ranges is needed.

Researchers have been working on the stability analysis of SCWR but a major impediment is the lack of reliable experimental dataset. Therefore, development of scaled down test facilities is essential for the study of instabilities under laboratory conditions. Comparison of the transient behaviour of SCWR and scaled fluid at SC pressure is very important in this downscaling. Moreover, a unified scaling methodology, applicable to downscaling of natural as well as forced circulation systems, is also desirable. It is evident from a meticulous survey of the relevant literature that, while natural circulation based systems have received some attention, supercritical channels with forced flow also experience a significant density variation across the core and hence are susceptible towards thermohydraulic instabilities. Therefore, detailed stability analysis of forced circulation SCWR is a crucial requirement for designing better control systems and safety measures. If an SCWR is subjected to an earthquake, the oscillating acceleration attributable to the seismic wave may cause the variation of the coolant flow rate, which might result in core instability due to the density-reactivity feedback. While large number of studies have been carried out to ensure the structural integrity of nuclear power plants during earthquakes, very limited studies have been reported on seismically induced thermal hydraulic instabilities in nuclear reactors, and none of them is focussed on SCWR.

Accordingly, the following are set as the objectives for the present thesis work.

1. To develop a new unified scaling methodology for natural circulation and forced circulation supercritical test facilities.
2. To develop a reduced order transient mathematical model of SCWR for thermal-hydraulics with and without coupled neutronics.
3. Analysis of instabilities and nonlinear dynamics of SCWR using the mathematical model.
4. Stability analysis of SCWR due to the seismic effects.

In the present work, a lumped parameter based approach is followed, visualizing the channel to consist of two distinct zones, separated by the pseudocritical point. The lumped parameter model (LPM) is derived from the basic governing equations of mass, momentum and energy. The basic one-dimensional governing equations, which are in PDE form, can be reduced to a system of ordinary differential equations (ODEs) by nodalization and spatial integration. LPM being computationally inexpensive, is suitable for detailed parametric studies of stability trends. The integrations of the conservation equations are done over the first and the second zones. The resultant system of algebraic and ODEs is employed for both linear and nonlinear analysis.

The scaling of SCWR system is done primary to identify a less restrictive model fluid, which can properly mimic the SCW under the relevant scaled condition of an SCWR, and to define the scaling rules in a generalized way, to preserve the phenomenological physics. US reference design of SCWR is considered as the prototype [Buongiorno, J., and MacDonald, P. E. Supercritical Water Reactor (SCWR) Progress Report for the FY-03 Generation-IV R&D Activities for the Development of the SCWR, In: U.S. Report No. INEEL/EXT-03-01210, INEEL, 2003]. Accordingly, the scaled dimensions of the lab-scale facility and corresponding operational settings in terms of power, flow rate, and inlet temperature are proposed. The advantages of the proposed methodology over the existing ones are stressed upon, along with a discussion on the role of involved dimensionless groups. It is found that the system pressure is independent of model fluids. R-134a is found to be the more suitable type of fluid among the fluids considered. The mass flux and power requirement are lesser which is significant for designing test facilities. Overall, the new proposed scaling method can be used to scale down all other parameters in both natural and forced flow systems.

For stability analysis, two different boundary conditions are taken into consideration, namely the constant pressure drop and constant mass flow rate. These can be viewed as the two extreme conditions of the pump characteristic curve. For analyzing the behavior of the system, two separate studies, by considering these utmost conditions, are carried out. A forced flow channel is simulated in the present study, with uniform heat flux on the wall. The channel is subjected to either constant pressure drop or constant mass flow rate boundary condition. The one-dimensional conservation equations for mass, momentum, and energy are taken. For the sake of generalizations, governing equations are nondimensionalized using suitable dimensionless parameters. The equations for mass and energy are integrated over the first node, i.e., from the inlet till the appearance of the pseudocritical boundary. Correspondingly, two ODEs are obtained in terms the location of pseudocritical boundary. Now, again integrating the conservation equations over the second node, from the pseudocritical boundary to the channel outlet, two ODEs are obtained in terms of the outlet enthalpy. These four equations are sufficient for the mass flow rate boundary condition. However, for pressure-drop boundary condition, the equation for conservation of momentum is integrated separately over both the zones and added to represent the net pressure drop across the channel. The system of six ODEs is used for analysis, along with algebraic equations which are obtained by equating the ODEs of mass and energy equations for the two zones, and the equation of state. In order to capture the sharp variation in thermophysical properties of supercritical water around the pseudocritical point, the equation of state is replaced by fitting a separate rectangular hyperbola for each of the zones [Pandey, M., Kumar, C.N., 2007. Lumped parameter modelling and stability analysis of supercritical water cooled reactor, in: The 12th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-12), Pittsburgh, U.S.A., 2007].

In order to introduce the power dynamics into framework, mathematical model for point reactor kinetics and a lumped parameter representation of the energy balance across the fuel rod are taken into consideration. The heat generated through nuclear fission can be described by point reactor kinetics, which is facilitated here by employing one group of delayed neutron precursors. The power generated in the reactor core being directly reliant on the neutron density, the non-dimensional equations govern the power and precursor density. The reactor kinetics is coupled with the fuel rod dynamics and thermal hydraulics through the reactivity feedbacks due to the fuel temperature and coolant enthalpy. Setting the steady-state reactivity to zero, the feedback reactivity can be expressed in terms of the temperature coefficient of reactivity and the enthalpy coefficient of reactivity for the corresponding fluid node. Here, the feedback at any time instant is considered to correspond to the average fuel rod temperature and average coolant enthalpies at each node at that particular instant. The heat generated inside the fuel rod is convectively transferred to the coolant from the rod surface. Applying energy balance with lumped capacitance approximation for the fuel rod, the state equation for fuel temperature and coolant enthalpy can be expressed in non-dimensional form.

Thus, the reduced order model obtained comprises a set of algebraic equations and ODEs, with dimensionless time as the sole independent variable. The model identifies the location of the pseudocritical boundary, exit enthalpy, exit mass flux, total pressure drop, fuel rod temperature , the power level and the precursor density as the six state variables, which can adequately represent the dynamic and stability behaviour of the system from thermalhydraulic as well as neutronics points of view. The thermalhydraulic part is compared with RELAP and it is found that the LPM results are more conservative than those predicated by RELAP, which implies that the use of LPM for stability analysis is safe. In addition to this, stability map for the various geometric and neutronic parameters are generated. It is found that increasing the channel length and hydraulic diameter has, respectively, destabilizing and stabilizing effects on the system. Increasing the fuel time constant and enthalpy reactivity constant has destabilizing and stabilizing effects, respectively. Moreover, five different mass flux profiles are analysed, which are used in the inertial pressure drop. It is found that the linear profile fitted throughout the channel is more suitable based on complexity of formulations and computational cost.

In order to examine the adequacy of the two zone model, a three zone model is also developed for stability analysis. The whole flow region is divided as heavy fluid region (H-F), heavy & light fluid mixture region (H-L-M) and light fluid region (L-F), following the approach proposed by Zhao et al. [J. Zhao, P. Saha, M.S. Kazimi, Hot-channel stability of supercritical watercooled reactors-I: steady-state and sliding pressure startup, Nucl. Technol. 158 (2007) 158–173]. After analysing the results, only 2–5% increases in the stability boundary is observed, which does not justify the complexity of the formulations. Hence, it is recommended that, for preliminary stability analysis, the two zone model is preferable over the three zone model.

Linear stability of SCWR channels have been studied in past. However, the analysis is valid only for infinitesimally small perturbations. Therefore, there is a need to carry out stability analysis for finite sized perturbations. The present thesis work has also attempted linear, as well as nonlinear, stability analysis of the channel and a specific boundary condition. A bifurcation analysis is carried out to capture the nonlinear dynamics of the system and to identify regions in the parameter space for which subcritical and supercritical bifurcations exist. The analysis shows that a generalized Hopf point exists in all the stability maps obtained. The subcritical and supercritical Hopf bifurcations are confirmed by numerical simulation of the time-dependent, nonlinear ODEs for the selected points in the operating parameter space. The identification of these points is important because the stability characteristics of the system for finite perturbations are dependent on them. Both stable and unstable limit cycles are detected and the boundaries of the unstable limit cycle are also calculated. It is found that, by increasing the length area ratio, mass flow rate and fuel time constant, the system is destabilized, stabilized and destabilized, respectively. The stability maps obtained by using the two boundary conditions, namely, constant pressure drop and constant mass flow rate, are compared with the stability maps based on a typical pump characteristic curve. The results are found to be qualitatively similar with minimal quantitative difference.

As pointed out earlier, it is important to properly evaluate the effect of the seismic acceleration on the core stability from a viewpoint of plant integrity estimation. In this study, the in-house code using LPM two zone model is employed for stability appraisal under the seismic acceleration, considering the one-dimensional neutron-coupled thermal hydraulic equations. The coolant flow in the core is simulated by introducing the oscillating acceleration attributed to the earthquake motion into the momentum equation as external force terms. A simple model based on sinusoidal acceleration obtained from the response analysis to the El Centro seismic wave of Imperial Valley earthquake is considered [Hirano, M., Tamakoshi, T. An analytical study on excitation of nuclear-coupled thermal hydraulic instability due to seismically induced resonance in BWR, Nucl. Eng. Des. 162, 307–315, 1996]. An artificial earthquake record with a nonstationary Kanai–Tajimi model is also used for more accurate simulation of the earthquake. Finally, the behaviour of the core and coolant are analysed in terms of various parameters of acceleration. Stability maps are plotted and the effects of the neutronics, amplitude, and direction of the oscillating acceleration are discussed.