Development of Mechanistic Modeling Capabilities for Local Neutronically-Coupled Flow-Induced Instabilities in Advanced Water-Cooled Reactors

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1. Research Objectives

The major research objectives of this project included the following;

- Formulation of flow and heat transfer modeling framework for the analysis of flow-induced instabilities in advanced light water nuclear reactors such as, boiling water reactors (BWRs) and supercritical water reactors (SCWRs)
- Formulation of a general multifield model of two-phase flow, including the necessary closure laws
- Development of neutron kinetics models compatible with the proposed models of heated channel dynamics; in particular, nodal and modal concepts for space-dependent kinetics
- Formulation and encoding of complete coupled neutronics/thermal-hydraulics models for the analysis of spatially-dependent local core instabilities
- Computer simulations aimed at testing and validating the new models of reactor dynamics.

A summary of the major research accomplishments is give below. Details concerning specific issues can be found in the publications listed at the end of this report.

2. Major research accomplishments

2.1. Formulation of flow and heat transfer modeling framework for the analysis of flow-induced instabilities in advanced light water nuclear reactors such as, boiling water reactors (BWRs) and supercritical water reactors (SCWRs)

The analysis of boiling channel dynamics has been conducted using various modeling frameworks and approaches, including both one-dimensional models and a multidimensional model of computational fluid dynamics. A summary of the individual models is given below.

<u>Models of Boiling Channel Dynamics</u>

The proposed models utilized both multifield (MF) and drift-flux (DF) modeling frameworks, both applied to two-phase steam/water flows. Multidimensional effects of two-phase flows have been accounted for using the multifield model, whereas the drift-flux modeling concept has bee used as a framework for a one-dimensional model of boiling channels in BWRs. The multifield model has been implemented in the computational multiphase fluid dynamics code, NPHASE-CMFD. On the other hand, a new stand-alone solver has been developed for a DF-model-based analysis of boiling channel dynamics.

<u>Space-dependent models of heated channel dynamics for fluids at supercritical pressures</u>

The Supercritical Water Reactor (SCWR) is one of several reactor design concepts included in the Generation IV International Advanced Reactor Design Program. This reactor design is based upon two well-developed technologies: current light water reactors and supercritical fossil-fuel power plants. Water at supercritical pressures is used as the reactor coolant. At these conditions, there is no phase change in the coolant, however the fluid properties undergo significant variation, particularly in the pseudo-critical region. For instance, the fluid density may decrease by a factor of six with increasing temperature. As it has been already observed for two-phase flow in boiling channels, variations in the fluid density may lead to density-wave oscillations, which in turn may cause many undesired problems for the system. It is important that the possibility of similar issues be addressed for flows at supercritical conditions because of the fluid property variations with temperature.

Two types of models have been used in the present analysis: a complete multidimensional CFD model which accounts for temperature- and pressure-dependent properties of fluids at supercritical pressures, and a simplified time-dependent and spatially-discretized one-dimensional model of heated-channel dynamics. The former model has been implemented in the NPHASE-CMFD, whereas the simplified model has been formulated in such a way that it can be readily encoded using any standard solver of ordinary differential equations.

2.2. Formulation of a general multifield model of two-phase flow, including the necessary closure laws

The multifield modeling concept has become a very popular approach to simulate multidimensional two- and multiphase flow and heat transfer. Although the multifield conservation equations seem to be a direct extension of those governing single-phase flows, it turns out that the averaging procedure introduces several constraints on the formulation of individual models.

One of the major issues in the development of the multifield modeling framework is a consistent formulation of the ensemble-averaged conservation equations of fluid flow and heat transfer, and of the associated models of interfacial phenomena between the continuous and disperse fields. The accuracy of computational predictions of gas/liquid two-phase flow and heat transfer strongly depends on the proper physical formulation of the governing interfacial phenomena.

An important aspect of modeling consistency is that dispersed-‘particle’/continuous-fluid flows do not follow the interpenetrating-media concept that is the foundation of the multifield modeling framework. However, it has been shown that the governing equations for such flows can nevertheless be converted into an approximate multifield formulation, provided that the continuous-field-induced pressure and shear stress for the dispersed field are equal to the respective terms for the continuous field.

Selected new and/or improved mechanistic models of gas/liquid interfacial forces have been introduced. In particular, a complete turbulence-induced interfacial force has been formulated, which is defined uniquely (i.e., without using arbitrary adjustable coefficients). This force is responsible for driving bubbles either away from the wall in both the central flow area and in the near wall region, although in each region a different force component plays the dominant role. Thus, the new force combines the roles of the commonly used turbulent-dispersion force and wall force. Also, a new multiple-bubble-size model of bubble/bubble interactions has been derived, consistent with the multifield modeling framework. The specific objective was to assess the effect of local bubble coalescence and breakup on the predictions of void fraction in gas/liquid two-phase flows.

Overall, a complete multidimensional Computational Multiphase Fluid Dynamics (CMFD) model of two-phase flow has been formulated, including local constitutive models applicable to two-phase flows in heated channels. The physical modeling assumptions have been parametrically tested and validated against experimental data. It has been demonstrated that the proposed model is capable of capturing various local flow and heat transfer phenomena.

2.3. Development of neutron kinetics models compatible with the proposed models of heated channel dynamics; in particular, nodal and modal concepts for space-dependent kinetics

A time-dependent one-dimensional two-group neutron diffusion model has been developed. This model has been first tested in a stand-alone fashion, and subsequently implemented in the NPHASE-CMFD code and coupled with the two-phase flow model discussed above. Test calculations of the combined thermal-hydraulics/neutronics model have been performed. The results obtained to date are very promising, although more testing is still necessary.

Another research direction has been concerned with the development of a modal model of reactor neutronics to account for the lateral interaction between parallel coolant channels. A modeling concept has been formulated and a numerical testing of the model has been performed in a stand-alone fashion. Also, the neutronics model has been combined with the model of fluid flow and heat transfer, as discussed below. Selected results of testing the coupled kinetics/thermal-hydraulics models are shown in Section 2 .5.

2.4. Formulation of complete coupled neutronics/thermal-hydraulics models for the analysis of spatially-dependent local core instabilities

Two formulations of the coupled neutronic/thermal-hydraulic couplings have been studied. First, a space and time dependent neutron diffusion model was coupled with a CMFD model of a reactor coolant channel. The coupling included the use of neutron flux to determine a local power distribution which was them used to evaluate thermal parameters of the reactor coolant (temperature, void fraction, volumetric fluxes). These thermal parameters were simultaneously accounted for in the evaluation of reactivity feedback effects on the neutron flux.

The other formulation was based on a modal approach to the spatial neutron distribution in a multiple parallel-channel core. In this case, a modal-series expansion was used for the lateral distribution of neutron flux between individual coolant channels (or groups of channels operating at similar conditions). In a separate development, the fluid flow and heat transfer model in a reactor coolant channel was coupled with a heat conduction model inside reactor fuel elements. Also, the thermal-hydraulic models of individual channels have been combined and used to formulate a model of multiple parallel channels subject to a common pressure-drop boundary conditions. This model was then combined with a modal model of reactor kinetics and used to investigate the channel-to-channel instability mode in Boiling Water Reactors (BWR).

2.5. Testing and validation stability of the models of reactor dynamics

(1) <u>Testing the time-dependent CMFD model of two-phase flow in boiling channels.</u>

Thew multidimensional model of combined flow and heat transfer in both single-phase and multiphase/multicomponent fluids has been extensively tested and validated. Figure 1 shows a comparison between the predictions of superficial velocities in a nonuniformly heated channel between the NPHASE-CMFD simulations and the predictions of the 1-D drift-flux model.

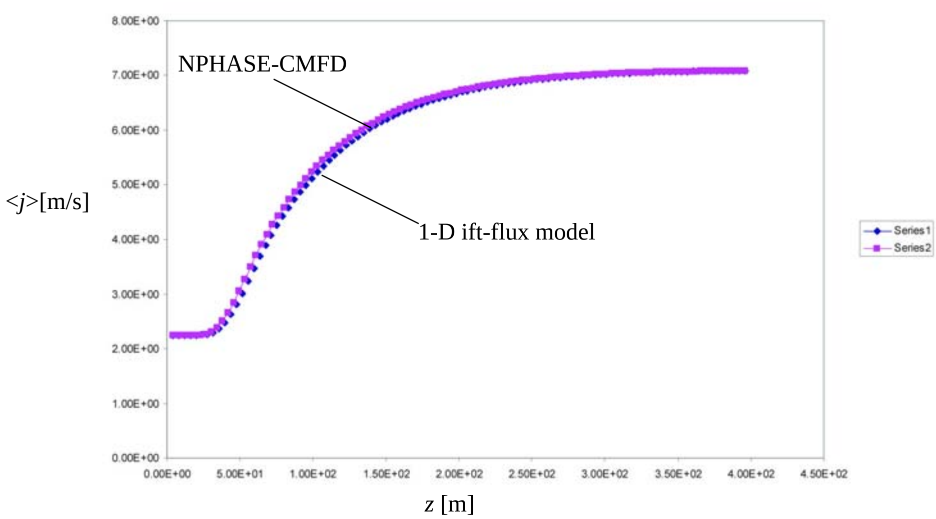


Figure 1. A comparison between the predictions of a multidimensional NPHASE-CMFD model and a 1-D drift flux model for the axial distribution of superficial velocity along a typical BWR coolant channel

(2) Testing the drift-flux model of a boiling channel

The assessment of the model's ability to predict the onset-of-instability conditions for the parallel-channel mode was then performed for the experimental conditions of Saha. To fully assess the consistency of the model, both a complete nonlinear model and its linearized version have been used in the comparisons. In the latter case, both the analytical frequency-domain analysis and direct time-domain integration were performed. As can be seen in Figure 2, the results obtained using three different methods agree very well with one another, thus confirming the consistency of the physical model formulation, as well as of the numerical temporal integration method.

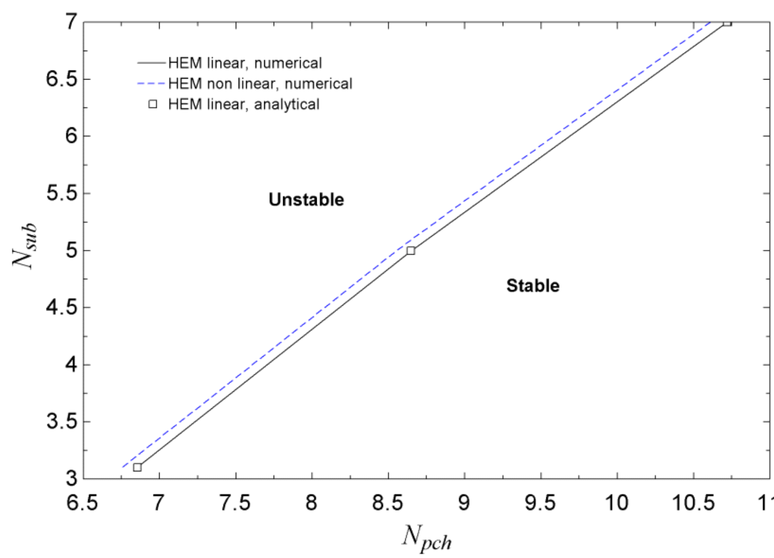


Figure 2. Neutral stability plots for linear and non-linear models, using direct temporal integration and a theoretical approach to the linearized equations.

(3) Testing of two-phase flow dynamics in multiple-parallel-channel systems

The model validation for the channel-to-channel instability mode has been performed using as a reference the experimental data of Aritomi. The experimental setup consisted of two identical parallel heated channels connected to common inlet and exit plena. Since the onset of instabilities for a system of two identical parallel channels corresponds to the conditions at which a single channel becomes unstable at a constant pressure drop boundary condition, the first step in the model validation process was to use the frequency domain method to determine the marginal stability conditions (in terms of the wall heat flux) for various assumed values of inlet velocity. As shown in Figure 3, the predictions agree well against the data. It should be mentioned here that because of a large water-to-steam density ratio at atmospheric pressure, the results of experiments data were very sensitive to the prescribed values of the operating conditions. Still, it was very interesting to notice that the theoretically predicted period of oscillations agreed very well with the measured value.

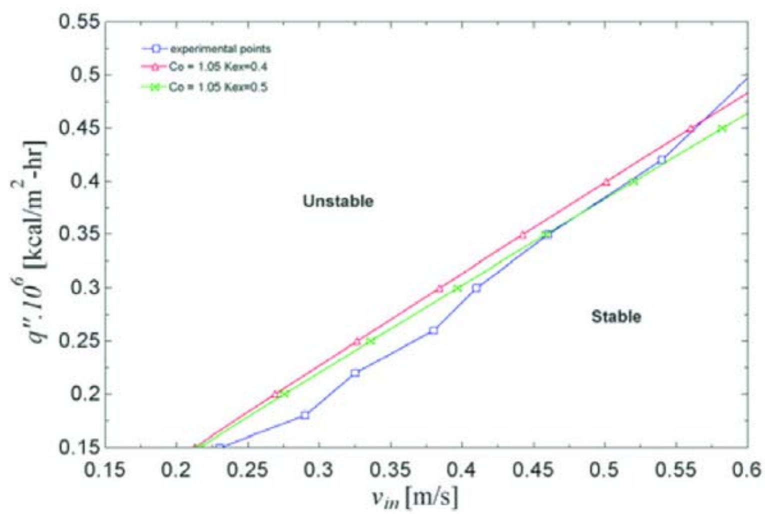


Figure 3. Theoretically-predicted stability boundary for two different values of the exit loss coefficient, compared against the experimental data of Aritomi et al.

Figure 4 through Figure 10 present the results of testing a model of three parallel channels. The geometry and power generation rates were identical in all three channels, whereas very small (1%) variations have been introduced in the local pressure loss coefficients at the inlet to individual channels. Figures 4 and 5 show long-term oscillations and details of oscillation patterns, respectively, for the individual channels for the case of initially perturbing the inlet flowrate to one of the channels. It is interesting to notice that the limit cycle oscillations in individual channels experience a 120^o phase shift, and small but significant differences in the magnitude of inlet flowrate oscillations. A comparison between Figures 4 and 6 shows that the long-term stable single limit-cycle oscillations are independent of the initial perturbation of the system, as expected. However, the initial period of time, which is required to approach the asymptotic stable limit-cycle model is much longer in the case of total flow perturbation (upstream of the parallel-channel system) that if the inlet flow rate in one of the channels is directly perturbed.

On the other hand, the oscillations in the common channel pressure drop, shown in Figure 7, experience period doubling, combined with some degree of asymmetry.

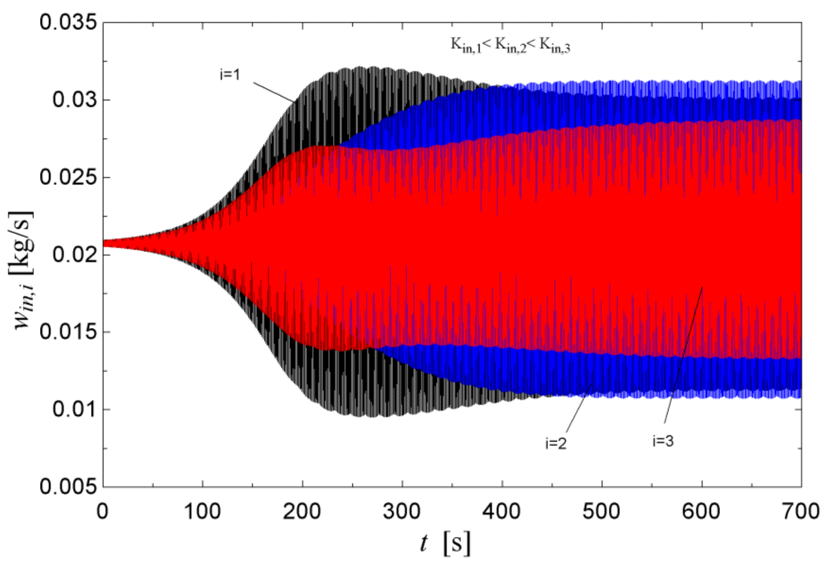


Figure 4. Limit cycle flowrate oscillations in a system of nearly-identical three parallel channels, having inlet loss coefficients different by 1%, in response to a flowrate perturbation at the inlet to one of the channels.

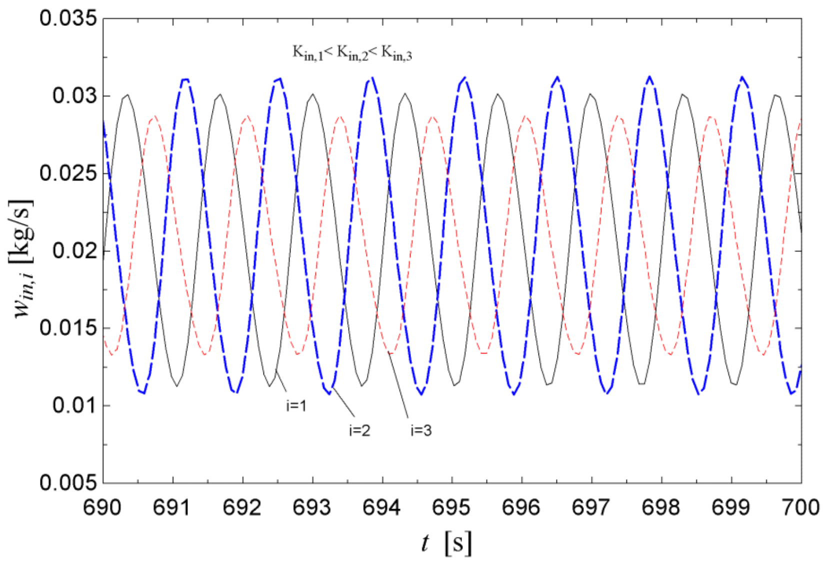


Figure 5. Details of well-established limit cycle oscillations in Figure 4, shown for an interval of 10 s.

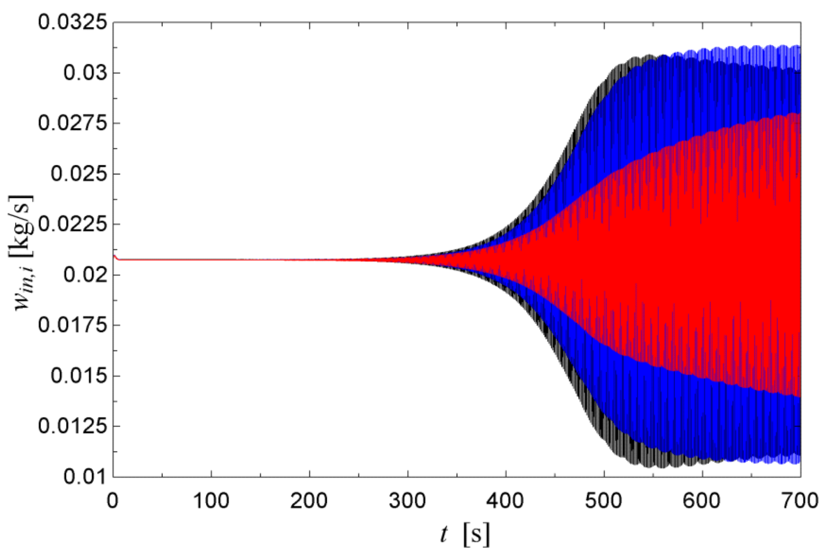


Figure 6. Limit cycle flowrate oscillations in the same system of nearly-identical three parallel channels as shown in Figure 4, in response to an initial flow perturbation in total inlet flow rate.

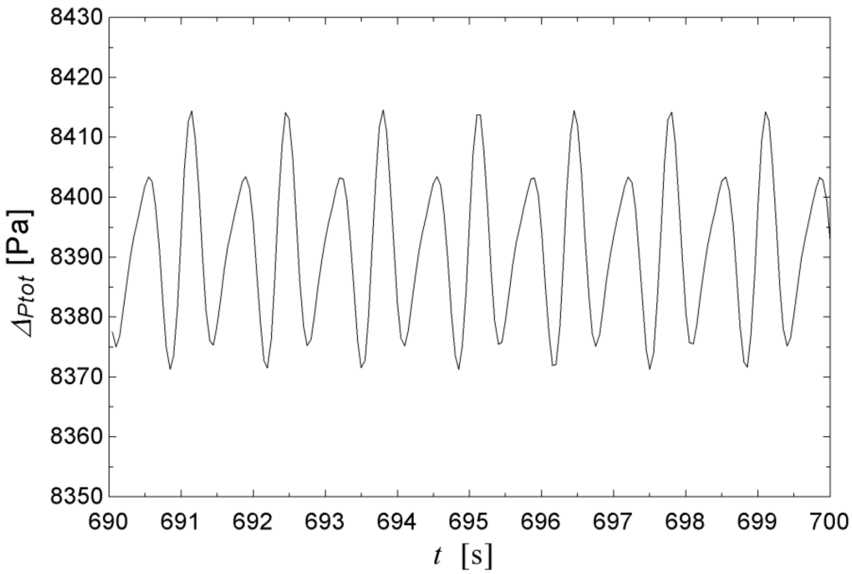


Figure 7. Pressure drop oscillations along the system of parallel channels shown in Figure 4, in response to an initial flow perturbation in total inlet flow rate. The mode of oscillations clearly shows period doubling phenomenon.

Figures 8-10 show the oscillation patterns for the same channels as those shown in Figures 4-7, but this time the inlet loss coefficients in two channels were varied by +5% and -5% with respect to the average channel. It is very interesting to notice a dramatic change in both the relative magnitude of oscillations and phase shift between the individual channels. Specifically, the channel with the lowest inlet loss coefficient. Channel-1, experiences the highest magnitude of oscillations and, furthermore, it oscillates out of phase with the other two channels, Channel-2 and Channel-3, which in turn oscillate in phase with each other. Naturally, the sum of magnitudes of flowrate oscillations in those two channels is equal to the magnitude of flowrate oscillations in Channel-1. As a result, the common pressure drop between the channels oscillates twice as fast as the channel flow rates, although a small effect of period doubling can also bee observed.

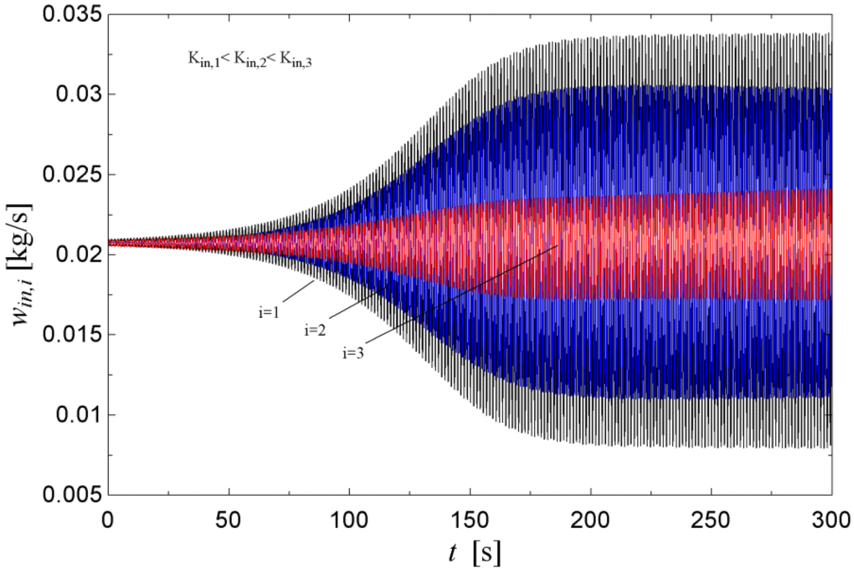


Figure 8. Limit cycle flowrate oscillations in a system of nearly-identical three parallel channels, having inlet loss coefficients different by 5%, in response to an initial flow perturbation in the inlet flow rate to one of the channels.

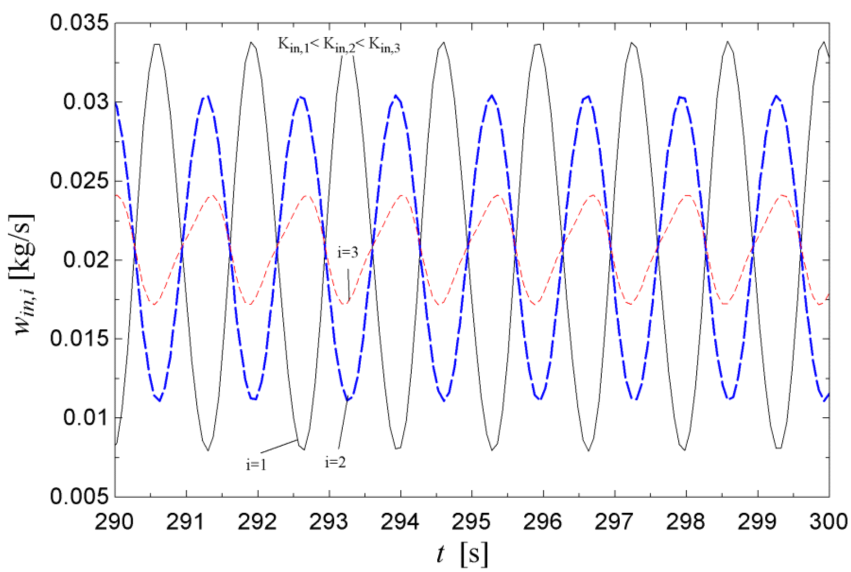


Figure 9. Details of well-established limit cycle oscillations in Figure 8, shown for an interval of 10 s.

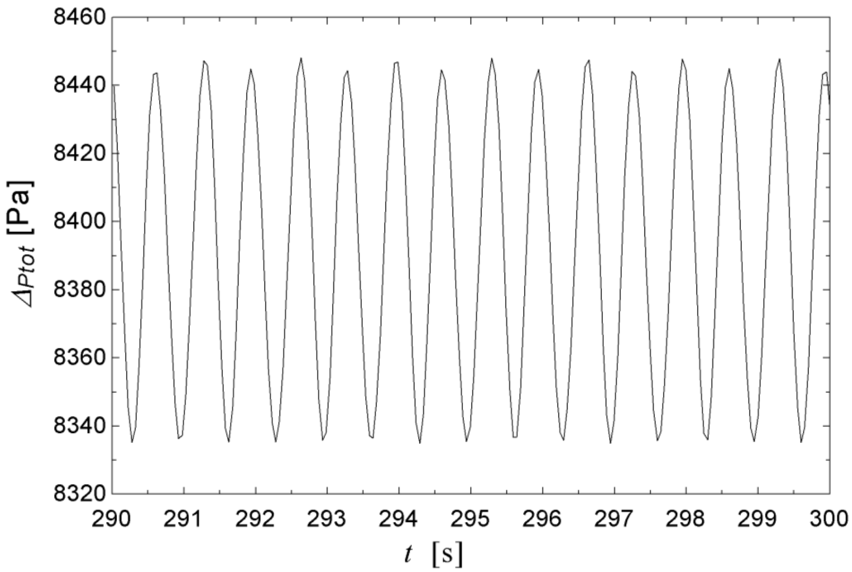


Figure 10. Pressure drop oscillations along the system of parallel channels shown in Figure 8, in response to an initial flow perturbation in the total inlet flow rate. The oscillations show a slight effect of period doubling.

(4) <u>Testing of the dynamics response of heated channel at SCWR operating conditions using a three-dimensional CFD model</u>

A multidimensional flow and heat transfer model in heated channels cooled with fluids at supercritical pressures has been implemented in the NPHASE-CMFD code and used to investigate the response of local phenomena to non-steady and transient flow conditions. The major features of the new model are: it includes complete models of variable properties of both supercritical water and CO_2, and it accounts for the thermal inertial of the heater. The model is applicable to both tubes with a central heated rod and to pipes with directly heated walls (annular heater geometry). Initial testing of the new model has been performed. In particular, simulations have been carried out for the transient response of a heated annular channel subject to changes in the wall heat flux and inlet mass flux. Figure 11 through Figure 15 show the time- and position-dependent fluid parameters in a heated channel following a sudden reduction in the pressure drop between the inlet and exit of the channel. A very interesting conclusion concerning the effect of variable properties can be drawn by comparing the results in Figures 11 and 12. Whereas the inlet mass flux decreases as a result of a pressure drop reduction, the fluid velocity at channel exit experiences an initial undershot, but later increases to a value higher that the initial velocity which corresponded to a higher pressure drop. This trend cab be readily explained by examining the density variations in Figure 14.

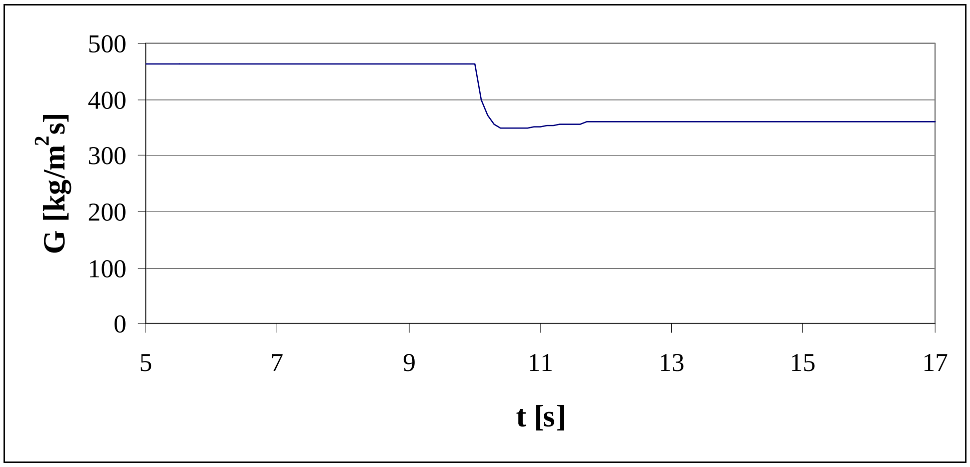


Figure 11. The dynamic response of a heated channel using superheated CO_2 as the cooling fluid to a sudden reduction in total pressure drop. The individual plots show the time history of fluid inlet mass flux at different axial locations.

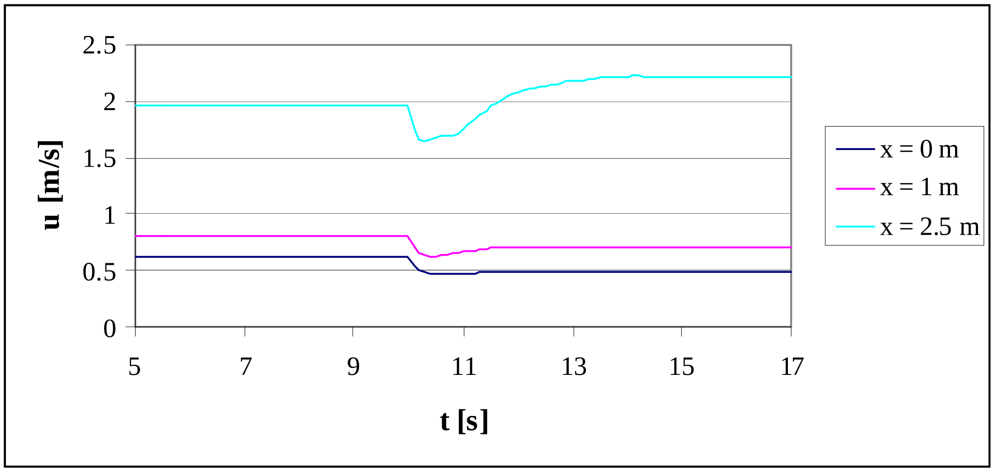


Figure 12. The dynamic response of a heated channel using superheated CO_2 as the cooling fluid to a sudden reduction in total pressure drop. The individual plots show the time history of fluid velocity at different axial locations.

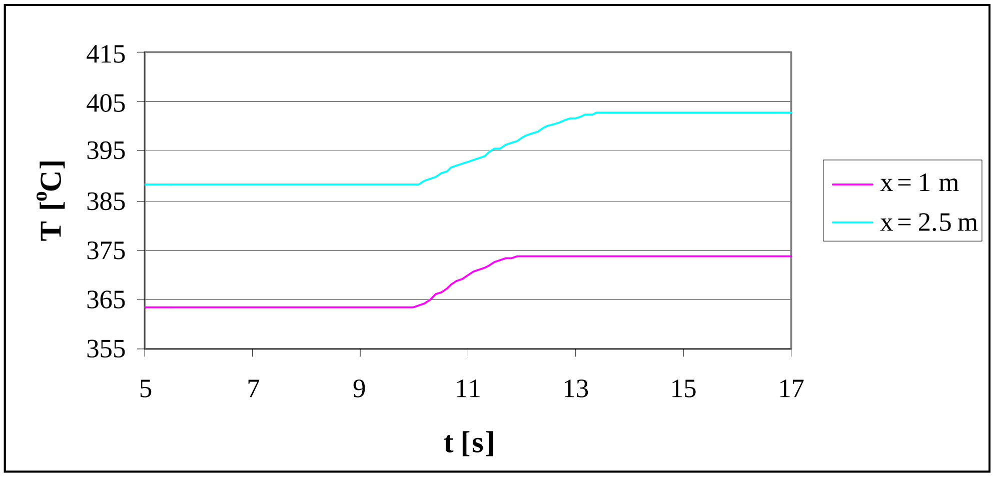


Figure 13. The dynamic response of a heated channel using superheated CO_2 as the cooling fluid to a sudden reduction in total pressure drop. The individual plots show the time history of fluid temperature at different axial locations.

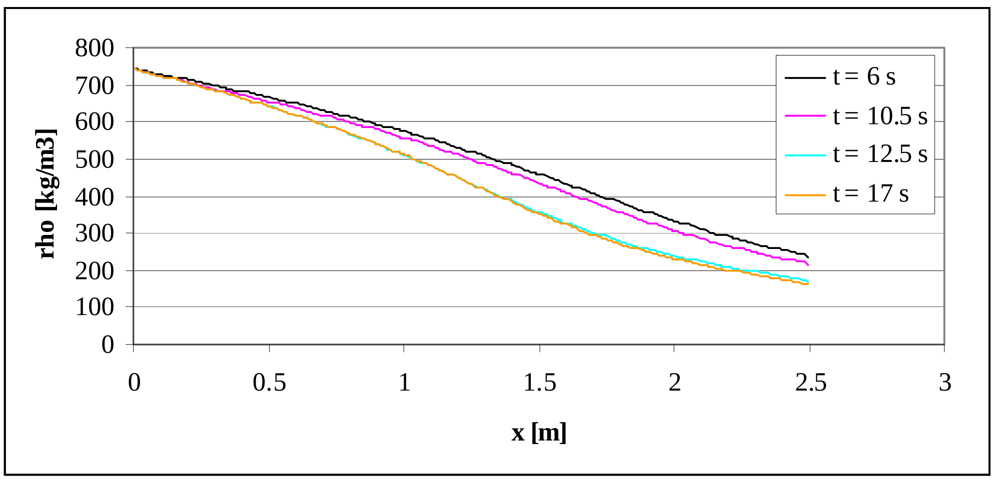


Figure 14. The dynamic response of a heated channel using superheated CO_2 as the cooling fluid to a sudden reduction in total pressure drop. The individual plots show the time history of fluid density at different axial locations.

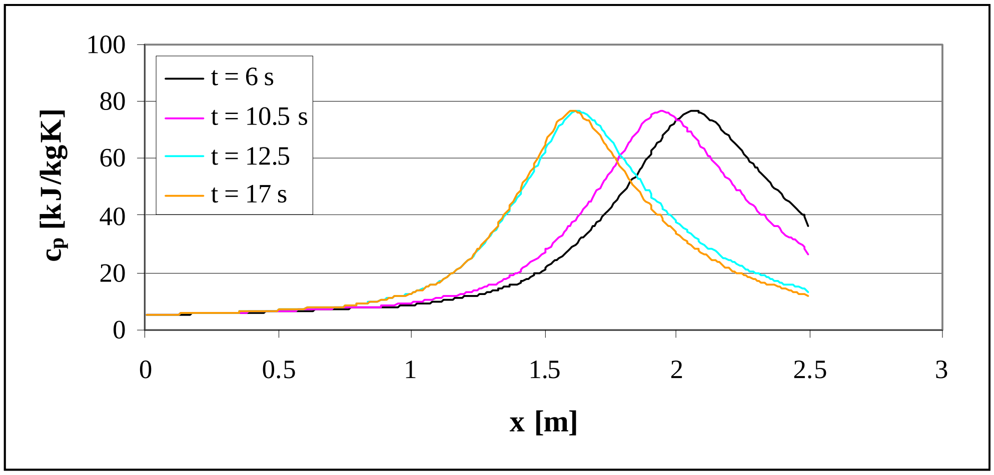


Figure 15. The dynamic response of a heated channel using superheated CO_2 as the cooling fluid to a sudden reduction in total pressure drop. The individual plots show the time history of fluid specific heat at different axial locations.

(5) <u>Stability analysis of SCWRs using a one-dimensional model</u>

As it was already mentioned in Section 2.1, although there is no phase change in the SCWR coolant channels, the fluid properties undergo significant changes, both spatial and temporal (see Fig. 15 above). Such variations in the properties of supercritical water can lead to density-wave oscillations, which can cause many undesired problems in reactor system operation. The purpose of the current study was to perform a one dimensional linear stability analysis, to better understand transient effects of local property variations on flow and heat transfer throughout the coolant channels of the proposed SCWRs. The parallel-channel instability mode was investigated. Specifically, the oscillatory response of heated channels subject to a constant-pressure-drop boundary condition was simulated using both time and frequency-domain methods. A grid dependency study was also performed to assure the numerical accuracy of the solutions. Then, a parametric study has been performed on the effect of channel operating conditions on flow oscillations at constant pressure drop boundary conditions. Typical results are shown in Figures 16-18.

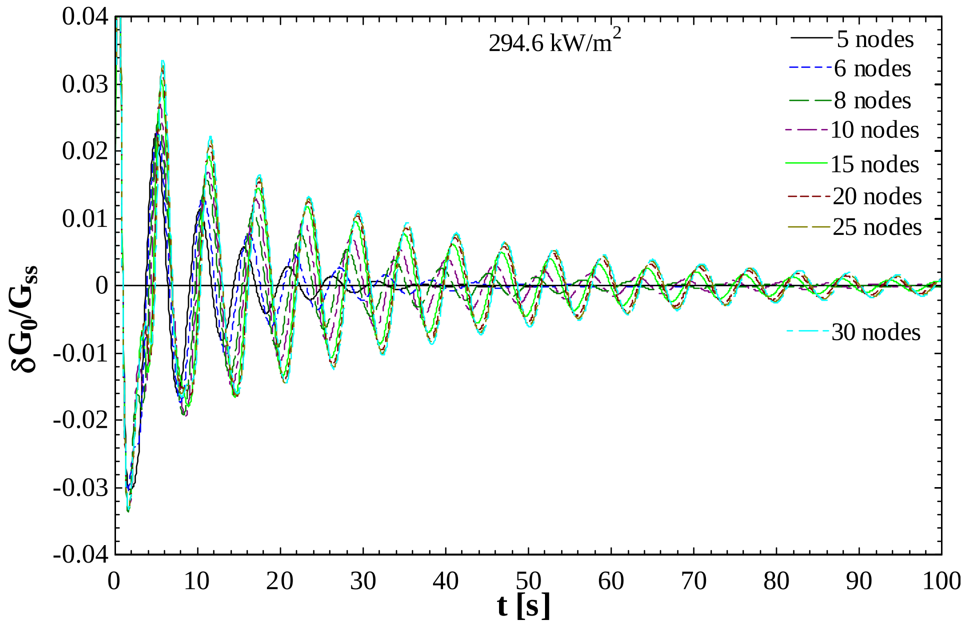


Figure 16. The results of a study on the effect of axial nodalization scheme on the oscillatory response of a stable coolant channel of a SCWR at P = 30 MPa.

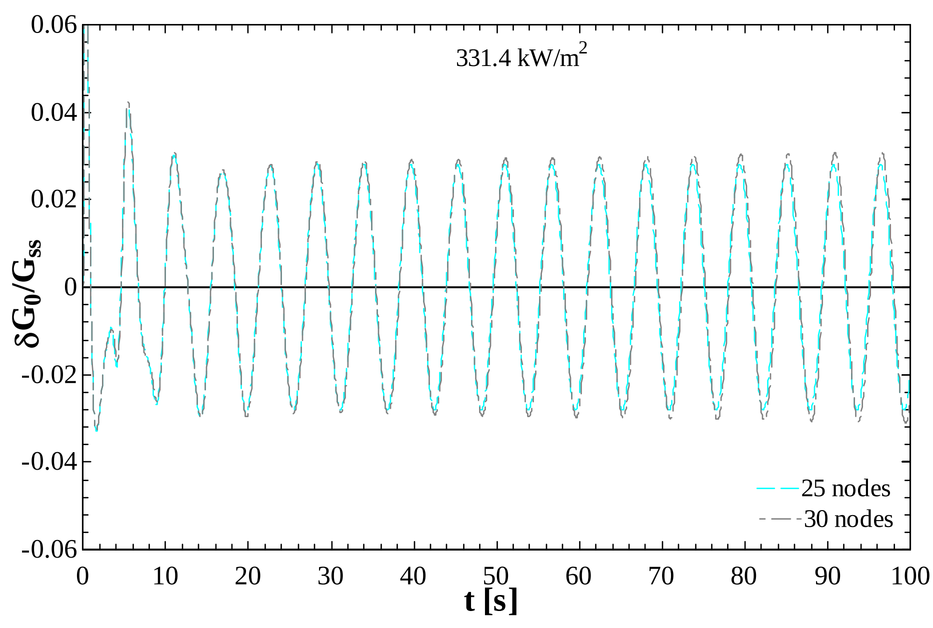


Figure 17. The results of a study on the effect on axial nodalization scheme on the response of a marginally stable coolant channel of a SCWR at P = 30 MPa.

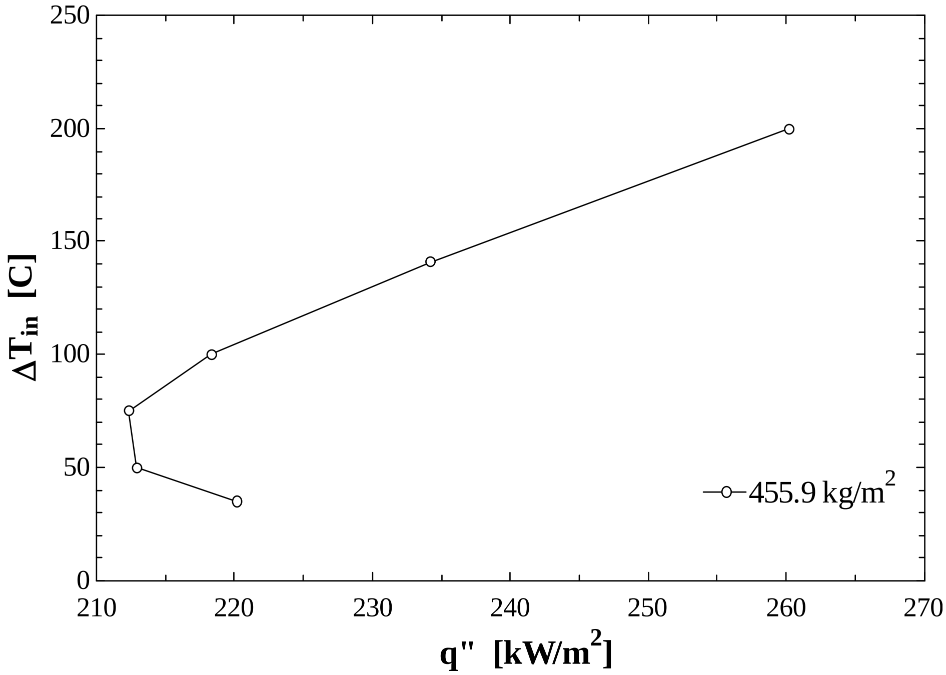
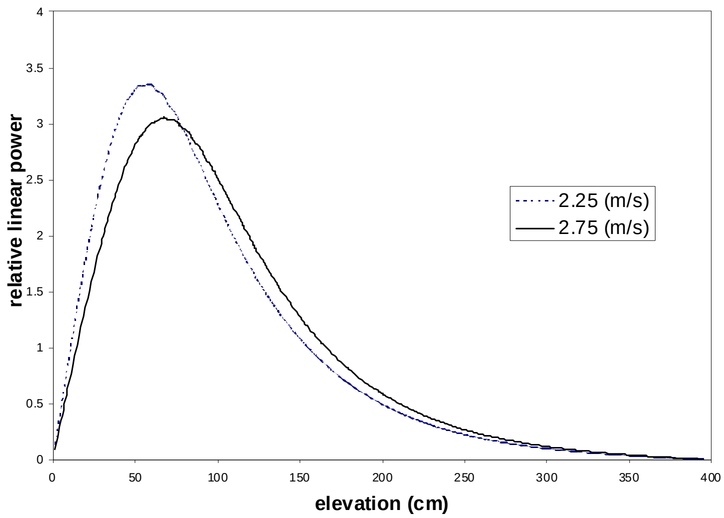


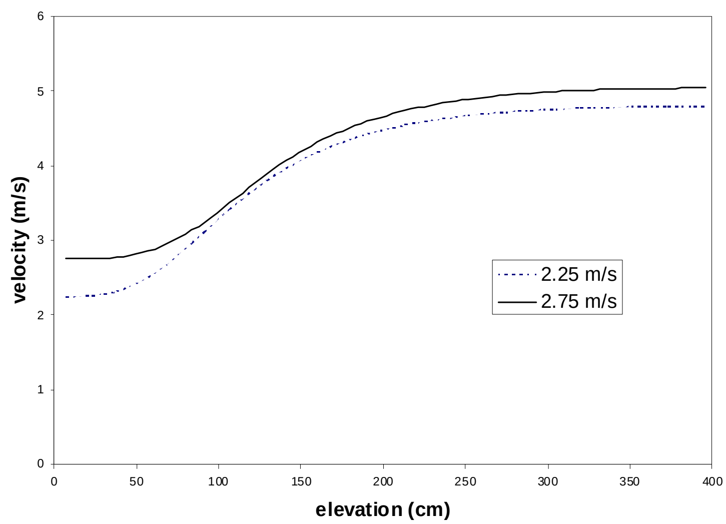
Figure 18. The stability boundary for the hot channel of a SCWR s at P = 30 MPa.

(6) Analysis of neutronically-coupled channel-to-channel instabilities in advanced boiling water nuclear reactors.

The major objective of the present work is to develop a consistent and computationally efficient model of the combined spatial/temporal neutron kinetics and core thermal-hydraulics, capable of predicting oscillatory transients in BWR systems in general, and typical modes of flow-induced instabilities in particular. Since both the two-group diffusion model and the various versions of the one-dimensional model of two-phase flow have been extensively studied before, the emphasis has currently been on testing the modeling consistency and numerical convergence and accuracy of the combined coupled neutronics/thermal-hydraulics model. Selected results of computer simulations are shown below. Figures 19 show the predictions of one-dimensional distributions of fuel heating rate and superficial velocity of two-phase flow for two different values of the inlet velocity, assuming the same total power in both cases. The mutual interaction between heating rate and velocity is due to the feedback effect between the void distribution and coolant/moderator cross-sections. As can be seen, the model properly predicts the effect of increasing void fraction on power distribution skewing (bottom-peaking), so that as the inlet velocity decreases the power peak moves closer to the inlet.



(a)

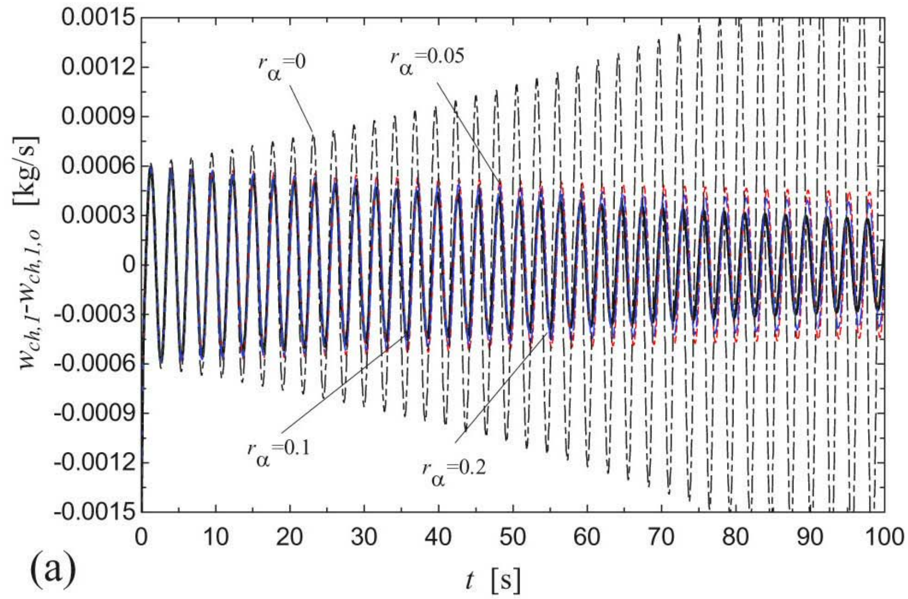


(b)

Figure 19. Axial distributions of axial power and volumetric flowrate distributions along a boiling channel of a boiling water reactor (BWR) for different inlet mass flow rates: (a) bottom peaked axial linear power distribution, (b) superficial velocity.

Another purpose of the present analysis was to investigate the fundamental aspects of the coupling between coolant dynamics in separate core zones on the onset of channel-to-channel oscillations. As it was discussed before, the proposed thermal-hydraulics model has been parametrically tested and validated against experimental data for electrically-heated channels. The fluid mechanics equations were then coupled with the neutron kinetics equations and used to study the onset of local channel-to-channel instabilities in a Boiling Water Reactor (BWR). It has been demonstrated that the model correctly predicts the marginal stability conditions, as well as the frequency of self-sustained oscillations between coupled parallel channels.

The simulations of neutronically-coupled oscillations in a BWR core confirmed that the void reactivity feedback plays a dominant role in the onset-of-instability conditions. In particular, as it is shown in Figure 20, the new model is capable of capturing the fact that increasing the magnitude of the negative reactivity coefficient beyond the range used in the current BWR may destabilize the reactor system.



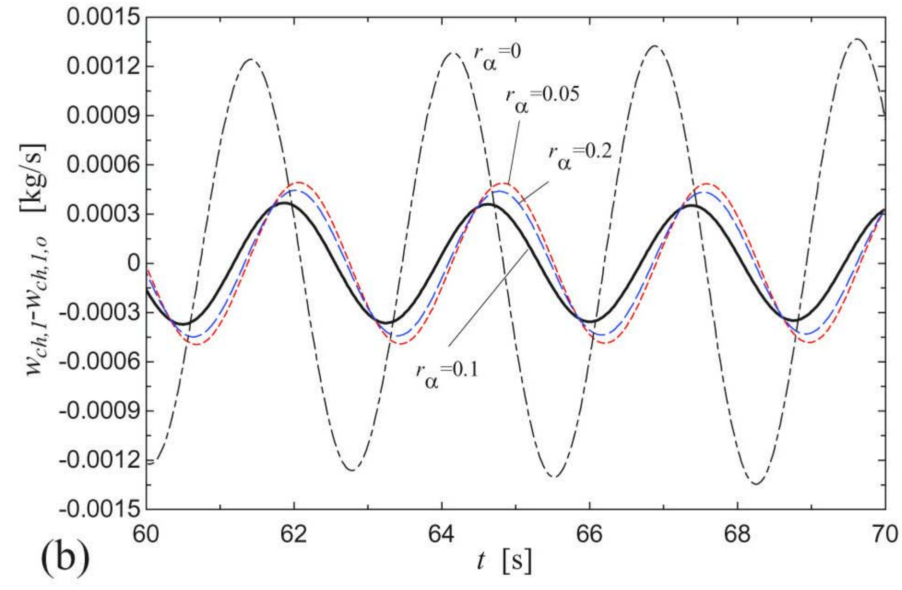


Figure 20. Parametric effects of void reactivity coefficient on reactor stability.

3. Conclusions and Plans for Work Continuation

Several issues of unsteady-sate thermal-hydraulics, core neutronics and combined thermal-hydraulics/neutronics coupling have been investigated in the present project. It has been shown that a multidimensional approach based on CFD concepts is capable of properly capturing local effects of flow and heat transfer dynamics in boiling channels and heated channel using fluids at supercritical pressures as coolants. Still additional work is still needed in order to develop fully-tested models of local effects in reactor dynamics and stability, in particular, the channel-to-channel oscillations in parallel channel cores of advanced light water reactors. It is important to mention that the research at RPI will continue on selected issues addressed in the project, at no cost to the sponsor. This is possible thanks to the fact that this project has been a part of PhD research requirements by two doctoral students who will still work on several unresolved issues needed for completion of their PhD programs. Finally, a complete list of publications documenting the results of the research on this project is shown below. It should be noticed that this list includes papers which have been submitted (and already accepted) to three international conferences planned after the termination of the project. This is why we have recently submitted a final request for a no-cost extension, to cover the travel expenses associated with attending those conferences and presenting our most recent results there.

4. Conference Participation and Presentations in 2009

1. Gallaway, T., “Multidimensional Analysis of Heated Channel Dynamics at Supercritical Pressures”, Presentation at 2009 ANS Student Conference, Gainesville, FL, April 1-4, 2009.

2. Gallaway, T., Antal, S.P. and Podowski, M.Z., “Multidimensional Modeling of Flow and Heat Transfer to CO2 at Supercritical Pressures”, Presentation at International Supercritical CO2 Power Cycle Symposium, Troy, NY, April 29-30, 2009.

3. Gallaway, T., Antal, S.P. and Podowski, M.Z., “Mechanistic Multidimensional Analysis of Heat Transfer in Fluids at Supercritical Pressures”, Presentation at the International Congress on Advances in Nuclear Power Plants (ICAPP’09), Tokyo, Japan, May 10-14, 2009.

5. List of Publications

1. Aniel-Buchhreit, S. and Podowski, M.Z., “On the Modeling of Local Neutronically-Coupled Flow-Induced Oscillations in Advanced Boiling Water Reactors”, Proc. ICONE-14 Conf., Miami, FL, 2006.
2. Gallaway, T., Antal, S.P., and Podowski, M.Z., “Multidimensional Model of Fluid Flow and Heat Transfer in Generation-IV Supercritical Water Reactors”, *Nuclear Science and Engineering*, 238, 2008, pp.1909–1916.
3. Podowski, M.Z., “Multidimensional Modeling of Two-Phase Flow and Heat Transfer”, International Journal of *Numerical Methods for Heat & Fluid Flow,* V. 18, Issue 3/4, 2008, pp.491-513.
4. Roubaud, S. and Podowski, M.Z., "On the Modeling of Channel-to Channel Oscillations in Boiling Water Reactors", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP’08), Anaheim, California, 2008.
5. Podowski, M.Z., "On the Consistency of Mechanistic Multidimensional Modeling of Gas/Liquid Two-Phase Flows", *Nuclear Engineering and Design*, V. 239, 5, 2009, pp.933-940
6. Gallaway, T., “Multidimensional Analysis of Heated Channel Dynamics at Supercritical Pressures”, Proceedings. 2009 ANS Student Conference, Gainesville, FL, April 1-4, 2009.
7. Gallaway, T., Antal, S.P. and Podowski, M.Z., “Multidimensional Modeling of Flow and Heat Transfer to CO2 at Supercritical Pressures”, Proceedings of International Supercritical CO2 Power Cycle Symposium, Troy, NY, April 29-30, 2009.
8. Gallaway, T., Antal, S.P. and Podowski, M.Z., “Mechanistic Multidimensional Analysis of Heat Transfer in Fluids at Supercritical Pressures” ICAPP’09, Tokyo, Japan, May 10-14, 2009
9. Roubaud, S. and Podowski, M.Z., On the Modeling of Channel-to Channel Oscillations in Boiling Water Reactors”, 13th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-13), Kanazawa City, Japan, September 27-October 2, 2009.
10. Gallaway, T. and Podowski, M.Z., “Stability Analysis of Fluid at Supercritical Pressure in a Heated Channel”, submitted to 2010 International Congress on Advances in Nuclear Power Plants (ICAPP '10), San Diego, California, USA, June 13-17, 2010.