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BOOK OF ABSTRACTS

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EVALUATION OF A NEW EQUATION FOR VARIABLE PROPERTY MIXED CONVECTION HEAT TRANSFER IN VERTICAL TUBES

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The physical and transport properties of fluids at supercritical pressure vary rapidly with increase of temperature as such fluids change from being liquid-like to gas-like over a particular range near to the so-called pseudocritical value. Consequently, when convective heat transfer takes place within them extreme spatial non-uniformity of fluid properties is present. Particular problems can arise as a result of the non-uniformity of density and onset of the influence of buoyancy. In some circumstances partial laminarisation of the flow occurs, accompanied by severe deterioration of heat transfer and localised overheating of the surface, which supplies heat to the fluid. Currently available equations for designing thermal systems in which such effects might be encountered are not able to take account of them reliably. Thus, there is a need for improved thermal design procedures. The author has recently been trying to develop semi-empirical models, which describe heat transfer in vertical tubes under conditions of buoyancy-influenced flow with strong non-uniformity of fluid properties. The aim is to construct a sound, theoretically-based framework which supports reliable procedures for calculating variable property, buoyancy-influenced heat transfer. Good progress has been made with this but a particularly important problem remains to be addressed, namely accounting for the inertia effects, which are present in such flows. Recently, this matter has been given detailed consideration and a promising approach identified. This is described in the present paper. Comparisons are presented between experimental data and calculations made using an equation, which incorporates the approach. Data for upward (buoyancy-aided) flow in vertical tubes and also for downward (buoyancy-opposed) flow under corresponding conditions of mass flow rate and heat flux are used. In each of the experimental investigations considered, measurements are available throughout the thermal development region and into that where a fully developed mixed condition is approached. The experiments provide a stringent test of the validity of the theoretically-based model and of the procedure used to account for the effects of inertia on thermal development. In the present paper the performance of the new equation in reproducing of observed behaviour is evaluated.

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INVESTIGATION OF THE EFFECTIVENESS OF JACKSON'S NUSSELT CORRELATION WITH BUOYANCY AND ACCELERATION TERMS IN CRITICAL WATER

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Jackson 2009 recently proposed a modification to a semi-empirical correlation for Nusselt number calculations specifically adapted for variable property heat transfer. To determine the effectiveness of this new approach in predicting supercritical heat transfer, experimentally measured heat transfer data was compared to that predicted by the correlation. The data used were from previous supercritical water experiments at the University of Wisconsin- Madison's supercritical water loop. The computer code EES which contains the NIST properties of supercritical water was used to calculate the predicted Nusselt number for all of the experimental conditions. Jackson's equation contains two coefficients relating to the acceleration and buoyancy parameters which have recommended values of 10000 and 2000 respectively, but which may be adjusted to better reproduce observed behavior. Calculations of Jackson's correlation were conducted where one coefficient was varied and the other held constant to best observe its effect and determine the best fitting coefficient for 957 data points where Gr/Re2.7 is greater than 1e-5, the conditions where heat transfer deterioration due to mixed convection are expected. The resulting calculated Nusselt values were compared to the experimental values to optimize the equation for the given data. It was found that the value of C1 (the coefficient related to thermally induced bulk flow acceleration) had little effect as expected because of the relatively large diameter (4.3 cm) used in the experiments. C1 was then chosen to keep the recommended value of 10000. C2, the coefficient for buoyancy had a much greater effect and was varied with the constraint that (C1Acb±C2Bob) FVP,1 stay below 0.382. With these conditions, C2 was optimized to approximately 475 for a maximum R2 value of a linear regression of the experimental vs. calculated Nusselt number.

MIXED CONVECTION HEAT TRANSFER TO CARBON DIOXIDE FLOWING UPWARD AND DOWNWARD IN A VERTICAL TUBE AND AN ANNULAR CHANNEL

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In development of an SCWR concept, the heat transfer at supercritical pressures is one of the most demanding research areas. Despite the numerous supercritical heat transfer correlations have been suggested in the past several decades, a search for a reliable and accurate correlation is still required, since the predictions from those correlations showed wide discrepancy each other. Especially in the deteriorated heat transfer regime, no correlation is able to produce accurate predictions. Under a strong buoyancy influence, the boundary layer structure is known to deform significantly, as the wall temperature approaches to the pseudocritical temperature. The deterioration may be a function not only of local flow properties but also possibly of flow history as well.

Some of currently available SCWR concepts introduce a multi-pass core design instead of single pass like the conventional LWR, in order to reduce hot channel factor. In the multi-pass design, coolant in at least one pass flows vertically upward. The flow direction is known to greatly affect a flow structure and results in completely different thermal characteristics between upward and downward flows.

This paper addresses three main subjects: 1) difference of thermal characteristics between upward and downward flows; 2) effect of simulating flow passage shape; 3) evaluation of existing supercritical heat transfer correlations. To achieve the objectives, a series of experiment were carried out with carbon dioxide flowing upward and downward in a circular tube with an inner diameter of 4.57mm and in an annular channel created between a tube with inner diameter of 10 mm and a heater rod with outer diameter of 8 mm. The hydraulic diameter of the annular channel based on a heated perimeter was 4.5 mm. The working fluid was CO2, which has been regarded as an appropriate modeling fluid to water, primarily because of their mutual resemblance in property variations against normalized temperatures. The mass flux ranged from 400 to 1200 kg/m2s and the heat flux was varied between 30 and 140 kW/m2 so that the pseudocritical point located in the middle of the heated section at a give pressure. The measurements were made at a pressure of 8.12 MPa. The pressure effect is known to be negligible and its effect, if any, is reflected through the property variation with pressure. The difference between the upward and downward flows was observed clearly. A heat transfer deterioration occurred over the wide range of enthalpy in the downward flow through the annular subchannel; however, the exact cause of that phenomenon is not clearly known and subject to further analysis. New heat transfer correlations were proposed. Several well known correlations were evaluated against the experimental data, and new correlations were suggested for both tube and annular channel.

CONVECTION HEAT TRANSFER OF CO2 AT SUPERCRITICAL PRESSURES IN VERTICAL SMALL, MINI AND MICRO TUBES

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Convection heat transfer of CO_2 at supercritical pressures in vertical small, mini and micro tubes with inner diameters of 2 mm, 1 mm, 0.27 mm and 0.0992 mm was investigated experimentally and numerically. The effects of heat flux, property variations, buoyancy and flow acceleration on the convection heat transfer were investigated. Detailed information generated by the numerical results, such as the velocity profiles and turbulence kinetic energy near the wall varying along the tube were presented to develop a better understanding.

The results show that for a small tube with an inner diameter of 2 mm, buoyancy was the dominant factor affecting the convection heat transfer rather than the flow acceleration. For cases with low inlet Reynolds numbers (less than 2500), buoyancy induces earlier transition from laminar to turbulent flow which increases the heat transfer coefficient. When heat fluxes are very high, the heat transfer in the tube is mainly controlled by the natural convection and the convection heat transfer coefficients for both upward and downward flows are very similar. For convection heat transfer in this small tube at relatively high Reynolds numbers (e.g. $Re_{in}\approx 9730$) and for the high heat fluxes, the convection heat transfer for upward flow is deteriorated by the strong buoyancy, while the convection heat transfer for downward flow is improved by the buoyancy.

For convection heat transfer in a small tube with an inside diameter of 1 mm at Reynolds numbers of 3300~5500, the effect of flow acceleration due to heating on the heat transfer for the experimental conditions is very weak, but buoyancy significantly influences the heat transfer at high heat fluxes.

For convection heat transfer in the mini tube with an inside diameter of 0.27 mm at low Reynolds numbers (less than 2900), the flow acceleration significantly influences the turbulence when the heating is relatively strong, and the buoyancy effect is relatively weak but still cannot be neglected. For convection heat transfer in the mini tube at relatively high Reynolds numbers ($Re_{in}\geq 4000$) for both low and high heat fluxes, the buoyancy and flow acceleration effects are insignificant. Numerical simulations using properly selected turbulence models accurately predict the convection heat transfer of supercritical pressure CO_2 in vertical small and mini tubes when buoyancy and flow acceleration are not significant.

For convection heat transfer in the micro tube with an inside diameter of 0.0992 mm, the local wall temperature varied non-linearly for both upward and downward flow when the heat flux was high. The difference of the local wall temperature between upward flow cases and downward flow cases was very small when other test conditions were held the same, which indicates that for super-critical CO_2 flowing in a micro tube as employed in this study, the buoyancy effect on the convection heat transfer could be neglected, and the flow acceleration induced by the axial density variation with temperature was the main factor that lead to the abnormal local wall temperature distribution at high heat fluxes.

Keywords: convection heat transfer; supercritical pressures; buoyancy; flow acceleration; upward flow; downward flow

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ANALYTICAL MODELLING OF THE HEAT TRANSFER TO SUPERCRITICAL WATER IN PIPE FLOWS

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The Supercritical Water-Cooled Reactor SCWR is investigated in order to achieve a higher thermal efficiency and to improve the economical competitiveness of light-water reactors. Due to the complex physical behaviour of supercritical water the heat transfer in the cooling channels may exhibit unusual behaviour such as heat transfer deterioration, with associated high temperatures of the cladding material. Among the various methods, e.g. empirical correlations, a lookup table or CFD codes to predict the wall temperatures of a heated channel or pipe the analytical method offers the advantage, that the influence of model parameters can be easily identified and the importance of physical mechanisms of flow and heat transfer can be understood as functions of the model parameters. Therefore, in the present work, an attempt is made to extend existing models for heat transfer in channel or pipe flows to variable-property fluids, i.e. supercritical water in a pipe. Emphasis is laid on the effect of the local maximum of the Prandtl number and the heat capacity near the pseudo-critical temperature on the laminar or turbulent heat transport.

The assumption of quasi-fully developed flow, Prantdl's mixing length eddy viscosity turbulence model, the two layer concept of a logarithmic wall-layer and a laminar sub-layer are analytically extended to a fluid with variable properties. The turbulence intensity of the flow is taken into account by a modified turbulence heat capacity, which is obtained by weighted averaging with a probability density function of the turbulent temperature fluctuations. For a given pipe radius, wall heat flux, mass flux and bulk enthalpy the heat- transfer coefficient and the wall temperature can be determined fully analytical. Mechanisms of heat transfer deterioration or enhancement can well be identified in the model.

Results are compared to various experiments of heated pipe flows at supercritical pressure, in particular Yamagata 72, Ornatski 71, Herkenrath 67, Shitsman 69, Bishop 68, and Griem 95. Good agreement is obtained in various cases with different types of heat transfer deterioration and enhancement. In parameter regions, where buoyancy, inlet effects, or a deviation from a quasi-fully developed state exists, the agreement is as expected poor. Therefore, the theory contributes to a better understanding of heat transfer of supercritical flows. It can also be used for the development of numerical wall functions, which are needed for an efficient numerical simulation using Computational Fluid Mechanics.

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CFD PREDICTION OF THE ONSET OF HEAT TRANSFER DETERIORATION TO SUPERCRITICAL WATER

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Supercritical water does not undergo phase change when heated and this is one of the reasons why it is considered as one of the preferred coolant types in future nuclear reactors. Currently a research program is undertaken in several countries to develop design principles for the supercritical-water cooled nuclear fission reactor. This type of coolant is also envisioned in fusion applications. Even though boiling, and thus boiling crisis, is not possible in systems with supercritical water, heat transfer to supercritical water may be significantly deteriorated under certain conditions. This phenomenon has been investigated for several decades, primarily through experimental studies performed in pipes. The experiments, as well as analytical work, indicate that the onset of Heat Transfer Deterioration (HTD) may be triggered by buoyancy effects in the thermal boundary layer, or by the property change in the viscous sub-layer. In this paper it is shown that both effects can be captured by careful computational fluid dynamics (CFD) analyses. In particular, for low flow conditions it is shown that the buoyancy effects dominate and the onset of the deterioration is due to density change of water in the boundary layer. CFD simulations capture the onset of HTD for vertical up-flow in a pipe and the predictions show very good agreement with measured data. For high flow conditions the onset of HTD is triggered by the thermal conductivity change of the supercritical water in the thermal boundary layer. This type of HTD is insensitive to the flow direction and to the presence of the gravity. It is shown that correct prediction of the onset of HTD requires specific resolution of the computational grid and a selection of adequate turbulence models. In current applications, correct results are obtained with grids that provide y+ less than 1 and when employing the Shear-Stress Transport (SST) model of turbulence. It is concluded that CFD is a reliable tool for prediction of the onset of HTD in pipes, provided the model is satisfying the above-mentioned conditions. Additional validation work is needed for CFD applications to such geometries as rod bundles with spacer grids.

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LIQUID-VAPOR PHASE SEPARATION DURING A SUDDEN QUENCH OF A SUPERCRITICAL FLUID

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We simulate liquid-vapor phase separation in a van der Waals fluid that is deeply quenched into the unstable range of its phase diagram. Our theoretical approach follows the diffuse-interface model, where convection induced by phase change is accounted for via a non-equilibrium (Korteweg) force expressing the tendency of the liquid-vapor system to minimize its free energy. Spinodal decomposition patterns for critical and off-critical van der Waals fluids are studied numerically, revealing the scaling laws of the characteristic length scale and composition of single-phase microdomains, together with their dependence on the Reynolds number. Unlike liquid-liquid phase separation of viscous binary mixtures, here local equilibrium is reached almost immediately after single-phase domains start to form. In addition, as predicted by scaling laws, such domains grow in time like $t^{2/3}$. Comparison between 2D and 3D results reveals that 2D simulations capture, even quantitatively, the main features of the phenomenon.

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NUMERICAL SIMULATION OF HEAT TRANSFER OF CO2 AT SUPERCRITICAL PRESSURES USING VARIOUS TURBULENCE MODELS

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This paper presents the numerical simulations of the convection heat transfer to carbon dioxide at supercritical pressure flowing through a mini tube with inner diameter of 0.27 mm performed using an 'in-house code' for axisymmetric heat transfer based on the Favre averaging approach. The numerical results are compared with the corresponding experimental data and the predicted values using the semi-empirical correlation for forced convection heat transfer of supercritical fluids.

The influence significance of the buoyancy and thermal flow accelerations varies with the heat flux for the in tube convection heat transfer to carbon dioxide at supercritical pressure at relatively low inlet Reynolds number (Re_{in}=2900) as reported in the previous experimental study. In another respect, the behaviors and respond to the modifications of the turbulence field due to influences of flow acceleration and buoyancy are important when evaluating the performance of the low-Reynolds number turbulence models in predicting the convection heat transfer of fluids at supercritical pressures, which give clues in future model improvement and development to predicted the flow acceleration and buoyancy affected heat transfer more precisely and in a broader range of conditions.

Three low-Reynolds number turbulence models, Launder-Sharma(LS), Lam-Bremhorst (LB) and V2F were employed in the present study to predict the in tube convection heat transfer to carbon dioxide at supercritical pressure for various heat flux (from 30 kW/m^2 to 113 kW/m^2) and various inlet Reynolds (Re_{in}=2900 and Re_{in}=5600) corresponding to the experimental study. A case with artificially increased heat flux (q_w = 226 kW/m^2) was also simulated in the present study.

Results show that for cases with relatively high inlet Reynolds number (Re_{in}=5600), in which the heat transfer were not significantly affected by buoyancy and flow acceleration, the LS, LB and V2F models are able to predict the local wall temperature distribution fairly well, as well as for cases with relatively low inlet Reynolds number (Re_{in}=2900) and low heat flux (q_w=30 kW/m^2). The difference among predicted results of various turbulence models is not remarkable. When the heat flux increased, the local wall temperature distribution was affected by buoyancy for upward flow, and both LB and V2F model responded to the buoyancy effect and reproduced the local wall temperature distribution trend but underestimated the heat transfer deterioration due to the buoyancy. The V2F model responded more strongly than the LB model. When the heat flux increased further, the flow acceleration shows more significant influence to the local wall temperature distribution than buoyancy effect. The predicted results by V2F model shows that the velocity profile was distorted by the strong flow acceleration effect, the turbulence was suppressed too much as a result, and the turbulence kinetic energy was nearly reduced to zero in the downstream of the test section. The LS model also over responded to the flow acceleration for higher heat flux cases, and the turbulence was suppressed by the distorted velocity profile which resulted in a local temperature distribution similar to the laminar results in the downstream of the test section. However, when increased the heat flux further artificially, the accumulated turbulence in the upstream due to the negative shear stress resulted from the distorted velocity profile and induced a heat transfer recovery in the downstream with the LS model.

Keywords: Carbon dioxide, supercritical pressure, turbulence models, buoyancy, flow acceleration, heat transfer deterioration.

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ASSESSMENT OF TURBULENCE MODELS IN THE SIMULATION OF HEAT TRANSFER TO WATER AT SUPERCRITICAL PRESSURE IN THE UPWARD AND DOWNWARD FLOW

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The paper summarises the results of joint work performed by the Universities of Pisa, Aberdeen and Manchester in assessing the performance of different CFD codes and turbulence models in the simulation of heat transfer experiments with water at an operating pressure of 25 MPa. The experimental data, obtained using a circular pipe test section, covered a broad range of conditions of heat flux, mass flux and inlet temperature for upward and downward flow.

Results, obtained by the SWIRL, the FLUENT and the STAR-CCM+ codes, showed that some models were effective in predicting heat transfer deterioration observed in upward flow for conditions where the wall temperatures were below the pseudocritical threshold. However, difficulties in predicting heat transfer phenomena quantitatively were evident when the pseudocritical temperature was reached within the boundary layer. The effects of buoyancy in deteriorating heat transfer in upward flow and enhancing it in downward flow are clearly demonstrated.

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EVALUATION OF HEAT TRANSFER COEFFICIENT OF SUPERCRITICAL WATER FLOWING INSIDE A TUBE USING CFD CODE FLUENT

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Although the heat transfer problem of pressurized supercritical water (SCW) flows in a round tube has been studied for decades, the subject is sill considerably of interest nowadays. This is partly because of the expanded investigation of using SCW for nuclear engineering applications like SCWR which is generation IV reactor and promising advanced nuclear systems because of their high thermal efficiency (i.e., about 45% as opposed to about 33% efficiency for current light water reactors LWRs) and considerable plant simplification. Literature survey shows that heat transfer coefficient (HTC) is sharply enhanced near the pseudocritical temperature. As the heat flux increases, the peak of the HTC decreases. When the heat flux reaches to some high values, heat transfer deterioration (HTD) occurs. CFD code with various turbulence models are being used to evaluate HTC. Modeling of Yamagata's experiment has been carried out for evaluation of HTC using CFD code FLUENT with standard k-ε turbulence model, non-equilibrium wall function, viscous heating, full buoyancy effect and including wall roughness effect. In this paper model constants for standard k-ε model have been derived. In the Yamagata experiment, investigations were made for HTC to supercritical water flowing vertically upward in vertical tubes of 10 and 7.5 mm internal diameter, at pressures 22.6, 24.5 and 29.5 MPa, bulk temperature from 230 to 540 ^{o}C, heat flux 233, 465, 698 and 930 kW/m^2 and mass flux 1200 kg/m^2.s. Two dimensional axisymmetry grid generation has been done using GAMBIT. Inbuilt boundary conditions in the FLUENT are invoked for mass flow rate at inlet, pressure outlet at the outlet of the tube and wall at the cylindrical surface where heat flux is given. Thermo-physical properties are taken from the (IAPWS IF-97) and piecewise linear variation are given in the FLUENT for 30 temperature points. Bulk fluid temperature is obtained using user defined function. HTC are obtained based on heat flux, surface temperature and bulk fluid temperature. The calculated HTC is compared with the experimental results and also compared with the results of the other authors. It is observed in both experimental and code calculated values that peak HTC decreases for increase in heat flux for constant mass flux and it is also noticed that peak HTC decreases with the increase in system pressure for constant heat flux. However, it is noticed that magnitude of peak HTC calculated by code is higher than the experimental data especially for higher heat flux and rate of decrease of peak HTC with increase in heat flux is lesser with compared to experimental results. It is observed that peak HTC increases with increase in wall roughness of the tube. It is also observed that HTC calculated by FLUENT code is in good agreement with the HTC calculated by other authors using CFD code with various turbulence models.

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REVIEW OF HEAT TRANSFER BEHAVIOR IN SUPERCRITICAL WATER COOLED REACTOR

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Supercritical water cooled reactor (SCWR) is one of the six reactor technologies under the US-led generation IV international forum (GIF). SCWRs are promising advanced nuclear systems because of their high thermal efficiency (i.e., about 45% as opposed to about 33% for current LWRs) and considerable plant simplification with a mission to generate low cost electricity. Various types of SCWRs i.e., pressure vessel type, pressure tube type etc. are under development stage. Literature survey showed that the majority of experimental data for measuring fluid temperature and wall temperature by varying pressure, mass flux, heat flux and geometry were obtained in vertical tubes, some data in horizontal tubes and annuli, and a few in other flow geometries including scaled down fuel bundles. But, actually fuel bundles are used in SCWR core. Therefore, more experimental data are needed for fuel bundles to verify the prediction of heat transfer coefficient. Empirical generalised correlations based on experimental data are obtained for heat transfer coefficient calculations at supercritical pressures. However, there is no consensus on the general trends in the predictions of heat transfer coefficient CFD codes are also used for the prediction of heat transfer coefficients. Wide variation in the prediction of the results is also observed for various turbulence models like standard k-ε, k-ω, RNG, RSM etc. In this paper heat transfer coefficient obtained by experimental results, empirical correlations and with the use of CFD code is discussed. In general, the experiments showed that there are three heat transfer modes of fluid at supercritical pressure: normal heat transfer, improved heat transfer and deteriorated heat transfer. Heat transfer at critical and pseudocritical pressures is influenced by the significant changes in thermophysical properties. Mechanism for enhancement/deterioration in heat transfer and criteria for onset of deterioration has also been discussed in this paper. Two important issues are identified which needs to be resolved for thermal-hydraulics/regulatory point of view: (a) heat transfer deterioration near pseudocritical temperature which leads to increase in wall temperature similar to dryout in LWRs, (b) fuel design criteria based on the critical heat flux is not applicable for the SCWRs because of no phase change of water inside the core.

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EFFECTS OF BODY FORCE AND VARIABLE PROPERTIES ON THE PERFORMANCE OF TURBULENCE MODELS

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The principal difficulty with computational prediction of heat transfer to fluids at supercritical temperature (SCPHT) using CFD is that the flow and turbulence fields can be significantly distorted under such conditions, which makes the prediction extremely difficult and poses a major challenge for turbulence modeling. Exploration of the full capacity of CFD in simulating supercritical pressure heat transfer has attracted great efforts from many researchers worldwide and good progress has been made. It is clear that carefully selected turbulence models can be used to satisfactorily predict many experimental data. Nevertheless, there are also cases especially when the heating is relatively strong, that no turbulence model can produce consistent and reliable results.

The proposed paper is aimed at investigating two particular aspects of the modeling of such complex flows. Firstly we will study the effect of non-uniform body forces on the modeling performance. Buoyancy as a non-uniform body force is a well-know difficulty in the modeling of mixed convection. The distribution of the body force in SCPHT problems is quite different from the mixed convection at normal pressure. Around the pseudo-critical temperature, there is almost a step-change in density which poses a further challenge. Investigations into this effect will be presented in the paper. Secondly, thermal properties of supercritical pressure fluids vary dramatically and nearly all models have taken no account of this. In fact, turbulence models often involve properties which may, in such flows, change so much to invalidate the predictions. This will be the second topic to be reported in the paper.

EXPERIMENTAL STUDIES ON CRITICAL FLOW AND HEAT TRANSFER OF WATER FOR NEAR-CRITICAL AND SUPERCRITICAL PRESSURES

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The experiments on critical flow, heat transfer coefficient and critical heat flux have been conducted at the test loop of supercritical water in China Institute of Atomic Energy (CIAE). The major characteristics and parametric trends of these phenomena are presented, and the experimental results are compared with the calculations of existing correlations and models.

For critical flow, more than 250 data points were obtained in two nozzles of 1.41 mm in diameter and 4.35 mm in length with rounded-edge and sharp-edge respectively, covering the ranges of inlet pressure of 22.1 – 29.1 MPa and inlet temperature of 38 – 474 K. The results show that in the near and beyond pseudo-critical region the thermal equilibrium is dominant, and the flow rate can be represented by the homogeneous equilibrium model reasonably. For the region of $0 < DT_{PC} < 100$ K the flow exhibits bifurcation behavior, characterized by a kind of instability, and the inlet shape of nozzle has a substantial effect on it. For the temperature well below the pseudo-critical point the flow is not at critical condition and is represented by the Bernoulli equation.

For heat transfer coefficient, the experimental data were obtained in a tube of 6 mm with upward flow, covering the ranges of pressure of 10 – 23 MPa, mass flux of 288 – 1298 kg/m^{2}s, local water temperature of 78 – 270 ℃, heat flux of 0.23 – 1.18 MW/m^2 and Reynolds number of 5.5×10^3 – 3.9×10^4. At both subcritical and supercritical pressures, the deterioration in heat transfer is observed at rather high Reynolds number as a result of the change in flow structure. The Dittus-Boelter type correlations e.g. Bishop's, Swenson's and Jackson's correlations, give better predictions of the heat transfer for the normal turbulent convection, but they can not predict the deteriorated heat transfer satisfactorily.

For critical heat flux (CHF), the data were obtained in a tube of D = 8 mm with water flowing upward, covering the ranges of pressure of 5–18 MPa and mass flux of 0.45 – 1.5 $\times10^3$ kg/m^{2}s. CHF increases with mass flux increasing significantly. It decreases distinctly when the pressure increases to the near critical region. The 96 CHF Look-Up Table gives reasonable prediction for higher flow, but substantial overprediction for mass flux lower than 1000 kg/m^{2}s.

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INVESTIGATION OF HEAT TRANSFER BEHAVIOUR IN TURBULENT, HORIZONTAL FLOWS NEAR THE CRITICAL PRESSURE

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Heating and Cooling mode heat transfer experiments were performed with supercritical carbon dioxide at several mass velocities and compared with numerical predictions from FLUENT. Studies utilized a type- 316 stainless steel, nine channel, semi-circular test section, and supercritical carbon dioxide serves as the working fluid throughout all experiments. The test section channels have a hydraulic diameter of 1.16mm and a length of 0.5m. Good agreement was found between the experiments and the simulations if the enhanced wall treatment was used with either the k-ε or the SST k-ω model. The SST k-ω model was preferred due to computational time, convergence and slightly better agreement with fewer mesh cells. The data and numerical model was also compared to several single fluid Nusselt number correlations and it was found that for bulk temperatures slightly away from Tpc ($0.98 < Tb/Tpc < 1.02$) the heat transfer coefficient agreed to within the error of the data. However, in near pseudocritical regions there is a lack of accurate prediction in heat transfer coefficient, with all correlations consistently over predicting experimentally determined values. Based on the horizontal flow orientation, buoyancy induced deterioration is likely not causing this effect, as the flow is at high Reynolds numbers, or low Gr/Re2 numbers. The reason for this over prediction is thought to be caused by an incorrect weighting of the specific heat terms. By studying the variation of the specific heat as a function of radial distance from the bulk to the wall, near the pseudocritical region, it is possible to determine the scaling factor for the changing properties. During this investigation it was also found that if the single phase correlations were evaluated at the film temperature rather than the bulk temperature the prediction was closer, however in all cases still overestimated the data in the near pseudocritical region.

CATHENA SIMULATION OF SUPERCRITICAL HEAT TRANSFER IN A TUBE

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This paper describes the application of the CATHENA MOD-3.5d/Rev 3 code to the steady-state flow in a heated pipe under supercritical conditions benchmark tests. The specifications for these simulations were defined under Task #9 of IAEA CRP "Heat Transfer Behaviour and Thermo-Hydraulics Code Testing for SCWRs". An objective of the IAEA CRP "Heat Transfer Behaviour and Thermo-Hydraulics Code Testing for SCWRs" is to test analysis methods for SCWR thermo-hydraulic behaviour and to identify code development needs.

The CATHENA code was developed by Atomic Energy of Canada Limited(AECL). The acronym CATHENA stands for Canadian Algorithm for Thermal hydraulic Network Analysis. The thermal hydraulic code CATHENA was developed primarily for the analysis of postulated upset conditions in CANDU reactors; however, the code has been qualified for a wider range of applications for the modelling of experimental thermal hydraulic test facilities. CATHENA uses a transient, one-dimensional two-fluid representation of two-phase flow in piping networks. An overview of the CATHENA thermal hydraulic code numerical methods and constitutive relations is given in Reference [1].

The CATHENA code includes thermophysical properties for both light water (H_2O) and heavy water (D_2O). The standard thermophysical property functions in CATHENA are based on functional fits [2] to the generating functions for H_2O and D_2O. For the latest code version (MOD-3.5d/Rev 3), an alternative set of light water property fits have been included extending their application into the supercritical pressure (to 100 MPa) region. The extended thermodynamic property functions for light water included in the MOD-3.5d/Rev 3 code version are based on the IAPWS generating functions [3], using the same fitting methodology. For the initial supercritical applications, simple extensions of the heat transfer and wall shear correlations from subcritical single-phase liquid conditions have been selected. For heat transfer the Dittus-Boelter correlation has been used. This study facilitates the inclusion of specific supercritical correlations in future code versions.

The benchmark problems described in this paper consist of steady-state flow of water in heated pipes at supercritical conditions. The first benchmark problem provides a comparison with experimental data from the open literature [4]. The other benchmark cases are upward and downward flow exercises where the data have not yet been published and will be used as "blind" simulation exercises. The results from the first benchmark exercise show that the simple extensions for supercritical flow used in CATHENA provide reasonable agreement for conditions towards the pipe inlet however the simulation results diverge from the experiment as the coolant temperature passes through the "pseudo-critical" enthalpy region further along the pipe. This level of agreement is expected given the simple heat transfer extensions used in this version of CATHENA.

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DESIGN PRINCIPLES AND FEATURES OF SUPERCRITICAL WATER-COOLED REACTORS TO MEET DESIGN GOALS OF GENERATION –IV NUCLEAR REACTOR CONCEPTS

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Research activities are currently conducted worldwide to develop Generation IV nuclear reactor concepts with goals of improving economic competitiveness, enhancing safety, improving sustainability, and enhancing proliferation resistance compared to modern nuclear power plants. Despite of the large effort, these goals had yet to be quantified (either numerically, physically, or practically) for each concept. The Super-Critical Water-cooled Reactor (SCWR) is one of the six Generation IV nuclear reactor concepts chosen for further investigation and development in several countries, including Canada, China, Japan, Russia, and European Union. At this point, there are two parallel SCWR designs; one based on the extension of the pressure-vessel type of reactors and the other on the pressure-tube type.

AECL has been focusing on the pressure-tube type of SCWR design synergetic to the existing fleet of CANDU reactors. A preliminary assessment has concluded that the pressure-tube SCWR design has the potential of meeting all design goals of the Generation-IV reactor concepts. The design process for the pressure-tube SCWR follows five logical steps that reflect successive knowledge refinement: Pre-Conceptual Design, Conceptual Design, Preliminary Design, Engineering Design, and Final Design. Each step in the process is defined as a discrete and identifiable improvement in the specification detail, reflecting increased feasibility, operating margin definition, refined safety limits, research maturity, and design completeness.

Options for the pre-conceptual CANDU SCWR design have been established and are in the process of being refined to optimize the design. The optimization focuses on meeting the design goals of the Generation-IV nuclear reactor concepts. Potential design-enhancement options have been examined and may be introduced in phases after the cost and benefit analysis.

This paper lays out the specific approach to achieving the design goals set by the Generation-IV International Forum (GIF). It introduces the disciplined design process leading towards the deployment of the CANDU SCWR. The major design features of the pre-conceptual CANDU SCWR are described and, where applicable, linked to the GIF Generation-IV reactor-design goals. Options for further enhancement of the design are identified.

THERMAL CORE DESIGN OF THE HIGH PERFORMANCE LIGHT WATER REACTOR

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The High Performance Light Water Reactor (HPLWR) is a SCWR concept, operated at an inlet pressure of 25 MPa with a core outlet temperature of 500°C. A thermal core design for this reactor has been worked out by a consortium of Euratom member states within the 6^{th} European Framework Program. Aiming at peak cladding temperatures of less than 630°C, including uncertainties and allowances for operation, the coolant is heated up in three steps with intermediate coolant mixing to eliminate hot streaks. Each fuel assembly is built from 40 fuel pins with 8 mm diameter and a pitch of 9.4mm, housed in a thermally insulated assembly box. Additional moderator water is foreseen in water rods inside each assembly and in gaps between the assembly boxes. With a thermal power of 2300 MW, a net electric power of 1000 MW shall be achieved, resulting in a net efficiency of 43.5%.

This concept has been studied with neutronic, thermal-hydraulic and structural analyses to assess its feasibility, which will be summarized in this paper. Coupled neutronic / thermal-hydraulic analyses by Maraczy *et al.* with the 2 group diffusion code KARATE and the one-dimensional code SPROD are defining an initial distribution of fuel enrichment, the positioning of the control rods, and the use of the burnable Gd absorbers to reach the envisaged power distribution. An equilibrium cycle analysis is showing radial form factors and the discharge burn-up. Different from conventional reactors, the radial power profile is intended to be non-uniform, with the highest power in the first heat up step in the core center and the lowest power in the second superheater step to result in the same peak cladding temperatures in each region.

Sub-channel analyses by Himmel *et al.* performed for different radial power gradients demonstrate the excellent coolant mixing inside assemblies thanks to the wire wrap spacers used in this design. Coolant mixing above and underneath the core has been studied by Wank *et al.* with CFD to design suitable mixing devices which reduce the temperature non-uniformities at the outlet of each heat up step. The analysis of natural convection phenomena in the gaps between assembly boxes approximated with a porous media approach by Kunik *et al.*, suggests changing the initial flow direction to an upward flow to reach stable moderator conditions.

A structural analysis of the assembly boxes by Herbell *et al.* led to a design concept for a thermal insulation, assuring sufficient stiffness with a honeycomb sandwich construction and with spacers between these boxes to minimize deformations during operation.

The paper gives an overview of the design concept and introduces to analyses performed by the consortium in this context.

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CURRENT STATUS OF RESEARCH AND DEVELOPMENT OF SUPERCRITICAL WATER COOLED FAST REACTOR (SUPER FAST REACTOR) IN JAPAN

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Pressure-vessel-type supercritical water cooled reactors (SCWRs) have been developed at the University of Tokyo (UT) since 1989 and are studied in Japan, Europe and other countries. UT's concepts of the reactor are termed "Super LWR" for thermal neutron spectrum and "Super Fast Reactor (FR)" for fast spectrum respectively. Fast spectrum option is expected to be possible with the same plant system as of the thermal option. The fast reactor will produce higher power rating than the thermal one with the same reactor pressure-vessel size, since the moderator is not necessary, so that the unit capital cost will be reduced further.

With the scope of developing an economical fast reactor system, a Japanese research project of the "Super FR" had been conducted since December 2005 to March 2010. UT, Kyushu University, Japan Atomic Energy Agency (JAEA) and Tokyo Electric Power Company (TEPCO) have participated in the project. It consisted of three subjects: (1) development of the Super FR concept, (2) thermal-hydraulic experiments and (3) materials developments. All items have successfully finished. The present paper will summarize the results of the project.

(1) Development of the Super FR concept

The purpose of the concept development is to pursue the advantage of high power density of fast reactors over thermal reactors to achieve economic competitiveness of fast reactors. The concept of Super FR was developed mainly by numerical simulations at UT. The studies covered fuel and core design, plant control, start-up, stability, plant heat balance, and safety analysis.

3D neutronic thermal-hydraulic coupled calculations were used for the core design. A candidate core was successfully designed with keeping both overall and local void reactivity negative by updating the fuel and core configurations. Solid moderators (ZrH layer) in blanket assemblies enables the Super FR to have a negative void reactivity without adopting flat core configuration or other special devices.

(2) Thermal-hydraulic experiments

To develop a thermal-hydraulic database for the design of the Super FR, thermal hydraulic experiments were carried out at Kyushu University and JAEA. The measurement of critical flow at depressurization, condensation of supercritical steam and the critical heat flux near the critical pressure have been made using a single tube and 7-rod bundles with surrogate fluid (HCFC22) at Kyushu University. Heat transfer tests with supercritical-pressure water were carried out at JAEA to verify the results obtained from the tests with HCFC22. A single heater rod and a 7-rod bundle including the grid spacer, which is the same as that in the Kyushu University for HCFC22, were tested. These results were used for validation of a CFD code, which was developed in this project, based on 3-D two-fluid model.

(3) Materials developments

Based on an advanced austenitic stainless steel (PNC1520) with high creep strength developed by JAEA for the sodium-cooled FBR, candidate materials for the fuel cladding of the Super FR were

manufactured. Tensile tests, creep tests, corrosion tests, slow strain rate (SSRT) tests, high temperature irradiation tests were conducted and showed good results.
Thermal insulations will be necessary for the Super FR due to large temperature difference in the reactor vessel. Requirements for the thermal insulator are low heat conductivity, low neutron absorption, and good thermal shock resistance and dimensional stabilities. As the candidate material, porous Yttria-Stabilized Zirconia (YSZ) of which heat conductivity was one-twentieth of Zirconia was developed.
A new approach was developed to evaluate elution characteristic of stainless materials in supercritical-pressure water. Radioactivated specimen (SUS304) with a known radioactivity is used as a sample and set at an autoclave vessel in supercritical water loop system. The elution database was accumulated using this approach.

Present study is the results of “Research and Development of the Super Fast Reactor” entrusted to the University of Tokyo by the Ministry of Education, Culture, Sports, Science at Technology of Japan (MEXT).

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SUPERCRITICAL WATER-COOLED NUCLEAR REACTOR (SCWR) CONCEPTS: THERMODYNAMIC CYCLES AND THERMAL ASPECTS OF PRESSURE CHANNEL DESIGN

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Research activities are currently conducted worldwide to develop Generation IV nuclear reactor concepts with the objective of improving thermal efficiency and increasing economic competitiveness of Generation IV Nuclear Power Plants (NPPs) compared to modern thermal power plants. The Super-Critical Water-cooled Reactor (SCWR) concept is one of the six Generation IV options chosen for further investigation and development in several countries, including Canada and Russia.
Water-cooled reactors operating at subcritical pressures (10 – 16 MPa) have provided a significant amount of electricity production for the past 50 years. However, the thermal efficiency of the current NPPs is not very high (30 35%). As such, more competitive designs, with higher thermal efficiencies, which will be close to that of modern thermal power plants (45 – 50%), need to be developed and implemented.
Previous studies have shown that direct cycles, with no-reheat and single-reheat configurations are the best choice for the SCWR concept. This paper presents several SCW NPP cycles based on direct, no-reheat and single-reheat regenerative concepts. The main parameters and performance in terms of thermal efficiency associated with these configurations is investigated in the first part of this paper.
There are a few technical challenges associated with the no-reheat and single-reheat SCW NPP configurations. The single-reheat cycle requires nuclear steam-reheat, thus increasing the complexity of the reactor core design. Conversely, the major technical challenge associated with a SC no-reheat turbine is the high moisture content in the LP turbine exhaust.
The SCWR core concept investigated in this paper is based on a generic pressure-tube reactor cooled with supercritical water. The considered reactor concept is based on a pressure-tube configuration with the following operating parameters: electrical power of 1200 MW, pressure of 25 MPa, reactor inlet temperature of 350°C, and reactor outlet temperature of 625°C.
In general, fuels currently investigated for the SCWR concept are high-temperature ceramics, similar to uranium dioxide (UO2). Previous studies have shown that if UO2 is used the centerline temperature of a fuel pellet might exceed the conservatively established industry accepted limit of 1850°C. As such, alternative fuel options, with higher thermal conductivities are being considered. The second part of this paper investigates a possibility of using uranium carbide (UC), uranium dicarbide (UC2) and uranium nitride (UN) as SCWR fuels since they have higher thermal conductivities when compared to conventional nuclear fuels such as UO2, MOX and thorium dioxide (ThO2).
Also, important safety parameters such as a bulk-fluid temperature, heat transfer coefficient, inner

sheath temperature and fuel centerline temperature have been calculated along the heated bundle-string length for non-uniform cosine-based Axial Heat Flux Profiles (AHFPs). To model a generic SCWR fuel channel, a 43-element bundle was assumed.

In addition, a new heat-transfer correlation for supercritical water flowing in vertical circular bare tubes was proposed. This correlation can be used for preliminary conservative estimation of heat transfer coefficients in supercritical water-cooled bundles, as bundle correlations have not been developed yet.

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DIMENSIONLESS PARAMETERS IN STABILITY ANALYSIS OF HEATED CHANNELS WITH SUPERCRITICAL FLUIDS AT IMPOSED HEAT FLUX AND WALL TEMPERATURE CONDITIONS

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The paper further explores the suitability dimensionless numbers proposed in past work for the analysis of flow stability in heated channels containing fluids at supercritical pressure. The achievements obtained by their use in the application to heated channels with imposed heat flux are summarised and the treatment is extended to imposed wall temperature conditions.

Among the aspects considered having relevance for analyses and experiments in support to the design of Supercritical Water Reactors, the following are given particular attention:

- basis for the selection of the dimensionless numbers;
- use of dimensionless numbers in defining stability boundaries;
- applications in fluid-to-fluid comparisons;
- dimensionless form to be adopted for temperature;
- identification of dimensionless numbers suitable for setting up heat transfer correlations with supercritical fluids.

The lines for further research in the field are finally discussed.

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STABILITY ANALYSIS OF GENERATION IV SUPERCRITICAL WATER REACTORS

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The Supercritical Water Reactor (SCWR) is one of several reactor design concepts included in the Generation IV International Advanced Reactor Design Program. This reactor concept capitalizes upon the experience gained to date in the technology of the current light water reactors and of the supercritical fossil-fuel power plants. In SCWRs, 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 [1,2]. In particular, the fluid density decreases by a factor of nearly six with increasing temperature. It has been seen for two- phase flow that variations in the fluid density can lead to density-wave oscillations, which may cause many undesired problems in system's performance [3]. Similar issues must be addressed for flows at supercritical conditions because of the fluid property variations with temperature. The stability studies for supercritical water systems which been performed before have been typically based on oversimplified models and their results are often not fully consistent and incomplete. The objectives of the present paper are twofold. First, the effect of local (multidimensional) property variations of fluids at supercritical pressures (such as water and CO2) and their impact on the dynamic response of heated channels is discussed.
Secondly, the methodology and results are shown of the analysis of density-wave oscillations in SCWRs using a complete one-dimensional model of reactor coolant channels.
The results of parametric testing and validation of the proposed model will be discussed in the full paper, including a sensitivity analysis to major modeling assumptions. These results include comparisons between time-domain integration of the governing equations, and the frequency-domain analysis using two different approaches to quantify the effect of axial distributions of: fluid properties, power distribution and transient heat transfer across fuel elements.
Newly developed SCWR stability maps will also be shown. They will be compared against similar results available in the literature.

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EXPERIMENTAL AND THEORETICAL INVESTIGATIONS ON STEADY STATE AND STABILITY BEHAVIOR OF NATURAL CIRCULATION SYSTEMS OPERATING WITH SUPERCRITICAL FLUID

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Supercritical water (SCW) is being considered as a coolant in some advanced nuclear reactor designs on account of its potential to offer high thermal efficiency, compact size, elimination of steam generator, separator & dryer, making it economically competitive. The elimination of phase change results in elimination of the CHF phenomenon. Supercritical water natural circulation loops are capable of generating density gradients comparable to two-phase natural circulation loops. SCW under natural circulation is also considered as a viable option for heat removal in some advanced nuclear reactor designs. Hence, the behavior of steady state natural circulation with supercritical fluids is of interest for a number of new reactor systems. Besides stable steady state, operation with unstable natural circulation is undesirable as it can lead to mechanical vibration of components and failure of control systems. Since SCW experiences drastic change in its thermo-physical properties (e.g. density) near the pseudo-critical temperature, natural circulation may be susceptible to density wave oscillations and ledinegg excursions.

A test loop has been set up in BARC to study steady state and stability behavior of natural circulation with supercritical fluids. The loop has been designed to operate with SCW as well as supercritical carbon-dioxide. The experiments have been conducted in the test loop with supercritical carbon-dioxide. The steady state and stability predictions made by in-house developed non-linear computer code (NOLSTA) have been compared with experimental data. The present paper describes the experimental results and analysis in detail.

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HYDRAULIC FEATURE INVESTIGATION ON SUPERCRITICAL WATER NATURAL AND FORCED CIRCULATION LOOPS

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A comparative investigation on steady-state hydraulic feature of both natural and forced water circulations under supercritical pressure is carried out. To extract the hydraulic essence and related features for these two types of water circulation under supercritical pressures, two simple yet typical loops of natural and forced circulation with almost the same geometries (except that the forced circulation loop is equipped with a pump) are considered for comparison. A unified hydraulic model for both natural and forced circulation is established, considering the unique variation of thermo-physical properties at supercritical pressures, along with the nonlinear momentum transport and system coupling characteristics. Related frictional pressure drop correlation, which has been developed under supercritical pressure conditions by other authors, is also included within the model to account for the special momentum transport characteristics in supercritical water flow. Furthermore, appropriate nonlinear numerical computation, which is based on continuation concept, is applied for numerical calculation for solving the model equations. Steady-state solutions of the model both for natural and for forced circulation present the complex but typical hydraulic features of both loop systems. A comparative study on static hydraulic features of both types of supercritical water circulations reveals the special non-monotonic relationships of mass flow-rate vs. heating power. Meanwhile, effects of such factors as inlet bulk temperature of heating section, local resistances and/or pump characteristics on system hydraulic and heat transmission features are delivered. Furthermore, characteristics of forced circulation driven by different real pumps and a ideal pump are compared.

Through the above-mentioned investigation, following conclusions are drawn, which deliver rather interesting and useful insights for future engineering application of supercritical water circulations and, for which further experimental validation is preferable:

(1) A specific feature of flow-rate - heat load relation of natural circulation under supercritical pressure is revealed. Flow-rate in the loop first increases and then decreases rapidly with heating power and reaches it maximum Gm at certain heating power Qm. Favorable operating region for local heat transfer and loop heat transport is found to be within range of Q < Qm. In contrast, mass flow-rate in forced circulation experiences first a nearly flat region followed with an abrupt drop one with heating power increases. Adverse condition in abrupt dropping region of mass flow-rate may possibly impair loop heat transport and operation safety.

(2) For natural circulation, both higher inlet temperature and local loss lead to flow-rate decreasing and make outlet temperature increase. Among the influences of inlet and exit local loss, the latter tends to be larger. For forced circulation, trends of inlet temperature and local loss effects on system hydraulic features are similar.

(3) In forced circulation, pump characteristics have obvious influences on system hydraulics and heat transport. Higher circulation flow-rate might be obtained through either increasing pump effective power or decreasing the slope of G - H curve. These strategies are also beneficial for loop heat transport.

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THERMOHYDRAULICS OF THE VVER-SCP SINGLE-PASS CORE HYDRO-PROFILING AND STABILITY

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At present OKB “GIDROPRESS” (Podolsk) and IPPE (Obninsk) are engaged in performing the conceptual design and computational studies of the single-loop reactor plant VVER-SCP cooled by supercritical water. Two variants of coolant flow arrangement in the core are considered: single-pass and double-pass. The paper emphasizes the thermohydraulic features of VVER-SCP single-pass core.

The computational studies performed for VVER-SCP core have revealed the necessity of the following:

– throttling of the fuel assembly inlet for the coolant flow thermohydraulic stability;

– hydro-profiling of the core for equalizing the coolant temperatures at the outlets of fuel assembles with different power;

– limitation of power peaking in fuel rods in the fuel assembly.

The paper gives a brief description of VVER-SCP design and the results of the computational studies and principal engineering solutions, aimed at assurance of coolant flow stability and equalization of coolant heating in different fuel assemblies, are presented. A brief description of the applied methods and codes is given.

PRELIMINARY NATURAL CIRCULATION DATA OF A SCALED SCWR EXPERIMENT

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The SCWR is one of the GEN IV designs currently under investigation. So far, a number of concepts have been presented in open literature for a SCWR: a Japanese concept, an American concept, a Korean concept and recently a European concept, labeled the HPLWR. This study considers this latest concept which is quite different from typical reactor cores, involving a three-pass flow layout with upper and lower mixing plena (see Fischer [1]) as well as a separate downward cold water stream inside the fuel assemblies for moderation purposes. As the GEN IV designs also focus on improved safety, using natural circulation is being studied extensively. Considering the large density difference in a SCWR (larger than in a BWR), natural circulation seems feasible. However, the large density difference is also expected to induce flow instabilities, as it does in a BWR.

In order to study the stability behavior of a natural circulation SCWR a setup was constructed. To alleviate temperature and pressure requirements of the setup, a scaling fluid (R23 at 57 bar) was used (using the scaling procedure presented by Marcel et al. [2]). The corresponding in and outlet temperatures for R23 are -21°C and 105 °C compared to 280°c and 500°C for water. The setup is a scaled version of the three-pass HPLWR design with a pre-heater section (moderation water) and a tall riser (10 m height in total) and is constructed of stainless steel tubing (6 mm ID for the core sections, 8 mm ID for the other sections). The heating (up to 18 kW) is done by sending a high current (up to 600 A) through the tubes. Local friction values can be imposed using valves at the inlet and exit of each core and at the top of the riser. Local fluid temperatures are measured at the inlet, and throughout the loop as well as the mass flow rate and local pressure drop values at the valves. At the top of the setup the heat input is extracted again using two heat exchangers (first one uses cooling water, second one evaporating R507a).

In this paper experimental natural circulation mass flow rate data will be presented. By increasing the heat input the mass flow rate in the loop initially increases but subsequently decreases again as the friction increases more rapidly with decreasing density. The impact of varying inlet temperature and local flow frictions was studied. Using a finite element code, the steady state behavior of the loop was simulated and these results were compared to the experimental data. Next to the steady state results, some dynamic experimental data will be presented showing the loop response to changes in the heating power. These results were compared to a dynamic finite element code.

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SUMMARY FOR A NUMERICAL SIMULATION ON A HPLWR FUEL ASSEMBLY FLOW WITH WRAPPED WIRE SPACERS AND RELATED WORKS

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The Supercritical or High-Performance Light-Water Reactor (HPLWR) is presently being investigated in Europe, Japan and other countries. In the cooling channels of the reactor core water at supercritical pressure of 25 MPa is strongly heated above the pseudo-critical temperature of 656 K.
For the understanding of the flow and heat transfer in the cooling and moderator channels of the core, CFD numerical simulation methods are applied. The coolant flow in a quarter fuel assembly with wrapped wire spacers was investigated in the previous work [1]. Approximate boundary conditions representative for a region within the evaporator of the so called Three Pass Core were chosen. The aim is to understand and model the transport of mass, momentum and heat between neighbouring sub-channels induced by the wrapped wire spacers and the increased flow resistance of the sub-channels. The geometrical model has been extended from a quarter to a full bundle. The boundary conditions have also been improved in order to omit the inlet effect. The latest results of this research will be presented in this paper.
The reference geometry was taken from Himmel et al. [2] with a wire revolution of 200 mm and the flow and thermal boundary conditions from Laurien et al. [3] in order to make comparisons to reference cases without wires. In the present ongoing work various computational cases with and without heating are under investigation.
Results and detailed interpretations of the flow and heat transport within the improved bundle model will be presented just like some previous validation and ongoing experimental works related to supercritical water heat transfer.

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INVESTIGATION OF THE FLOW AND HEAT TRANSFER OF FUEL ASSEMBLY IN SUPERCRITICAL WATER NUCLEAR REACTOR

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This paper is based on a CFD study of the flow and heat transfer inside the fuel bundles of a supercritical water nuclear ***fast reactor***. CFD is applied to the investigation of a 7-bundle fuel assembly of supercritical water reactor (SCWR) as depicted in Fig. 1. The numerical simulation of the flow and heat transfer inside the assembly channel produces the wall temperature distribution of the fuel rod along the perimeter direction. The amplitude of the wall temperature oscillation increases following an increase of the fluid's bulk temperature but the periodicity remains stable. Inside the assembly channel, the temperature and density of the fluid are also non-uniform. A secondary flow appears inside the channel and its intensity will increase following the increase of fluid bulk temperature. These investigations are significant for the future design of a SCWR.

Keywords: CFD, Supercritical Water Reactor (SCWR), Fuel Assembly

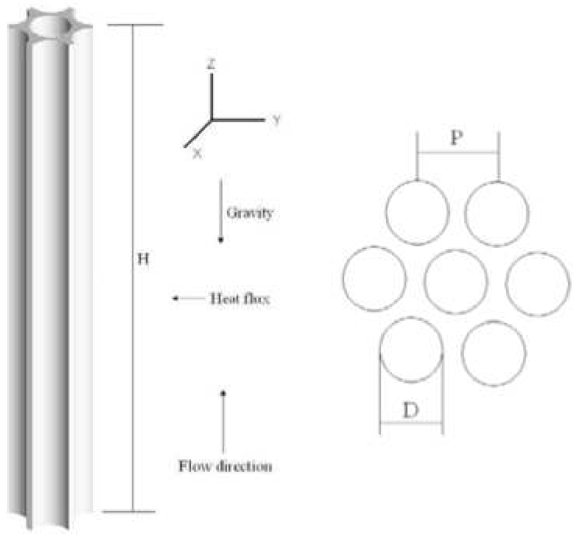


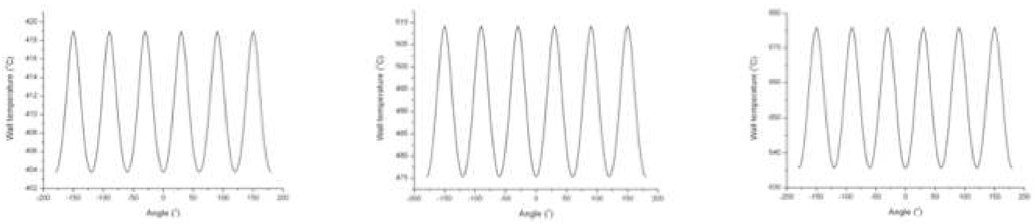
Fig. 1 Geometry of fuel assembly (D=7.0mm, P/D=1.16 and H=3000mm)

Main Results

Table 1 Calculation parameters

Mass flux（kg/m^2s）	Heat flux（kW/m^2）	Inlet temperature（°C）	System pressure（MPa）
1500	720	370	25

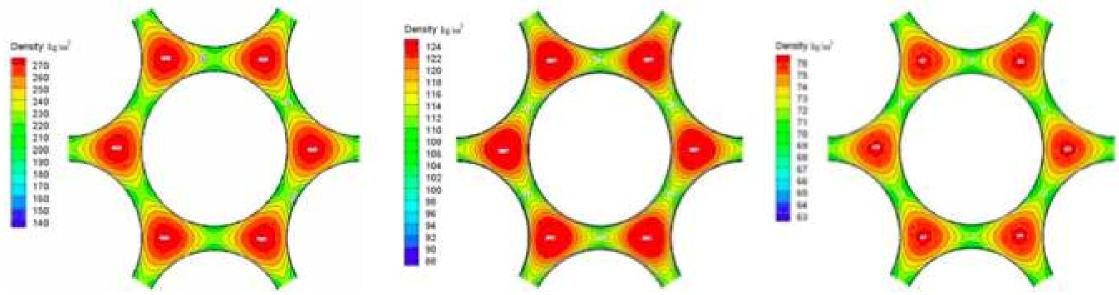
The flow conditions used in the present study are listed in Table 1. The wall temperature distribution along the middle rod perimeter is presented in Fig. 2, which shows that it varies periodically around the rod wall. The amplitude of periodicity and the mean temperature increased along the axial flow direction while the frequency remains fixed. The amplitude of the periodic temperature distribution is 15.2oC at the location Z = 900 mm, 34.0oC at Z = 1900 mm and 40.5oC at Z = 2900mm, respectively. Generally, a serious periodic distribution of the temperature on the rod wall will induce material deformation and damage to the rod due to the alternating thermal stress. This phenomenon is important for the design of SCWR.



Z=900mm Z=1900mm Z=2900mm

Fig. 2 Temperature distribution along perimeter at the middle rod wall

Presented in Fig. 3 is the density distribution inside the tunnel at three different locations. At Z = 900mm, the maximum density difference can reach 130kg/m3. However, near the outlet (Z=2900mm), the maximum density difference inside the tunnel is only 13kg/m3. The fluid density is mainly affected by the temperature because of the nature of the supercritical water. This phenomenon has to be considered in the nuclear physical calculation at the design stage. Especially at low temperature regions, a finer mesh has to be employed in the reaction cross-section calculation of nuclear physics to consider density effects.



Z=900mm Z=1900mm Z=2900mm

Fig. 3 Density distribution

Parts of conclusions

It is found from CFD simulations that the rod wall temperature distribution in SCWR is periodic and the maximum amplitude is about 40oC. This situation should be considered at the design stage. The density distribution is non-uniform. The near wall density is generally lower than the middle area inside the tunnel. This phenomenon will affect the nuclear physical calculation for the exact power estimation of SCWR.

SUBCHANNEL ANALYSIS OF WIRE WRAPPED SCWR ASSEMBLY

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Application of wire wrap spacers in SCWR can reduce pressure drop and obtain better mixing capability. As a consequence, the required coolant pumping power is decreased and the coolant temperature profile inside the fuel bundle is flattened which will obviously decrease the peak cladding temperature. The distributed resistance model for wire wrap was developed and implemented in ATHAS subchannel analysis code. The SCWR wire wrapped assembly was analyzed. The results show that: (1) the assembly with wire wrap can obtain a more uniform coolant temperature profile than the grid spaced assembly, which will result in a lower peak cladding temperature; (2) The pressure drop in a wire wrapped assembly is less than that in a grid spaced assembly, which can reduce the operating power of pump effectively. (3) The wire wrap pitch has significant effect on the flow in the assembly. Smaller H_{wire}/D_{rod} will result in stronger cross flow and a more uniform coolant temperature profile, and also a higher pressure drop.

SUPERCRITICAL WATER: ON A ROAD FROM CFD TO NPP SIMULATIONS

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The Fission and Radiation Physics Group at the Aalto University is contributing to the Finnish SCWR activities within the GEN4FIN-network [1]. Our research involves reactor core thermal hydraulics, and in particular, heat transfer phenomena in supercritical water including both theoretical studies and simulations with APROS [2] and OpenFOAM [3]. APROS is a software applicable to full-scale power plant simulations and OpenFOAM an open source CFD code. The complicated heat transfer in the supercritical region is a very challenging problem for the design of SCWRs and their safety assessment.

The steam tables of APROS have been extended to the supercritical region and their functionality has been tested with, e.g., blowdown simulations where the transient is rapid, hence mainly challenging for numerical stability whereas heat transfer has negligible effects. Numerous different heat correlations for supercritical water have been suggested, but simulations of benchmark experiments [4] have shown that for instance fuel clad temperatures generally cannot be described sufficiently accurately. This discrepancy has been encountered in several process simulation codes. The largest errors occur near the pseudo critical line, during the heat transfer deterioration. It turns out that the physics in supercritical water is clearly more intricate than in ordinary boiling heat transfer where rather satisfactory heat transfer correlations are available.

Full 3D CFD calculations allow a better description of various aspects of heat transfer in the supercritical region, i.e., effects arising from turbulence, buoyancy, varying material properties etc. On the other hand, CFD calculations are not feasible for plant-scale simulations. We have selected some simplified geometries and parameter ranges to study SCW heat transfer in a reactor. Old experiments [4,5] have been calculated with satisfactory results with OpenFOAM to check its validity. A steady state case of heat transfer in a circular pipe with upward flow was in agreement with OpenFOAM in contrast to simulations by APROS, which underpredicts wall temperatures by up to 200 degrees. In the future, different flow geometries will be studied with the main emphasis on fuel bundles. Heat transfer correlations for specific geometries would provide a remarkable improvement for the current situation. To obtain practicable results would require a general correlation which would stand against experiments.

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SOME PROBLEMS OF FLUID DYNAMICS AND HEAT TRANSFER IN SCWRs WITH ROD-BUNDLE CORES

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A brief survey of some problems appearing in designing rod-bundle Fuel Assemblies of supercritical water reactors' cores is presented. Those problems concern both getting appropriate coolability of core fuel elements, satisfactory neutron balance in the core and assurance of mechanical stability of fuel elements and other Fuel Assembly structures.
Different impacts of thermodynamic and kinematic properties of supercritical water and their fluctuations in area of pseudo-critical transition upon behavior of fluid-dynamics and heat transfer are considered and discussed.

It is shown incorrectness of application of the "heat transfer coefficient" concept to supercritical flow regions, where there is a strong dependence of water properties on enthalpy and pressure.
Different wide-spread turbulent models are considered from point of view of their applicability to simulating of supercritical fluid-dynamics in area of pseudo-critical transition.
Principle need of account of coolant compressibility as well as turbulent exchange by momentum and energy due to density fluctuations in numerical simulation of supercritical fluid dynamics is discussed.

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NEW SUPERCRITICAL WATER LOOP IN NUCLEAR RESEARCH INSTITUTE ŘEŽ, PLC – DESCRIPTION AND FIRST OPERATIONAL EXPERIENCE.

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In the frame of projects developing reactors under the framework of Generation IV International Forum, the need for new facilities for irradiation experiments has arisen. ÚJV has recently designed and built the Supercritical Water Loop (SCWL), an in-reactor experimental facility for irradiation testing of materials and water chemistry of SCWR systems, co-funded by the Structural funds of European Union and the European research project HPLWR Phase 2. The loop was designed for operation in the research reactor LVR-15 situated in ÚJV Řež (Nuclear Research Institute), Czech Republic.

The reactor LVR-15 is a tank type reactor with nominal power 10MW, fuel: IRT-2M with 36% enrichment, (IRT-4M with low then 20% enrichment from 2011) thermal flux $1.5x10^{18}$ n/m^{2}s and fast flux $2.5x10^{18}$n/m^{2}s.

SCWL shall serve for the purpose of corrosion testing of candidate materials, studies of radiolysis in supercritical water, water chemistry optimization and development of sensors for water chemistry monitoring. The maximum testing temperature and pressure in the testing zone of the irradiation channel are designed for 600°C and 25MPa, respectively. The loop shall provide data that will enable expansion of current knowledge mainly on materials performance from unirradiated to irradiated conditions and radiolysis model from sub-critical to supercritical conditions.

The loop enables dosing of gases (H_2, O_2, etc.), measurement, sampling and purification of the primary medium. The loop has been commissioned in fall 2008. Pressure tests and first functional out-of-pile tests were carried-out in 2009-2010.
The loop description, first experience with the loop operation at supercritical parameters and experimental possibilities are described.

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LOW TEMPERATURE CYCLES WITH SUPERCRITICAL FLUIDS FOR NUCLEAR PLANTS

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The supercritical power cycles are taking advantage of real gas behavior in order to achieve high thermal efficiency. The two most common supercritical cycles perform with water and carbon dioxide. The supercritical water cycle enhances thermal efficiency by rising turbine inlet temperature, while the supercritical carbon dioxide (S-CO_2) takes advantage of reduction of compressor input power due to properties change close to the critical point (30.98°C, 7.38MPa). The goal of this study is to check possibilities of designing power cycles in the range of lower temperatures with higher efficiency. There are indications that it might be possible to design conversion cycles with maximum temperature between 100 and 200°C with efficiency near that of the Carnot cycle. Supercritical fluids are the candidate optimal media for these cycles. The principle is to include certain parts with accelerated flow into the conversion cycle.

Thermodynamic analyses and comparison of different modifications of the S-CO_2 cycle in terms of cycle thermal efficiency have been performed.
An experimental S-CO_2 loop was built in 1999 in the Czech Republic. The main objective was to obtain experimental data for comparison with previous theoretical studies. This facility was the first of its kind in the world. Its operation and performed measurements have provided many interesting data and thus brought valuable operational experience as well as new objectives for future research and development of S-CO_2 cycles.

Main new factor to be included into cycles is heat transfer into accelerated flow, which can be mainly exploited into low temperature cycles. The detailed analysis summarize specific factors of this process.

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R&D ACTIVITIES AND PROGRAM ON SCWR IN NPIC

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This paper gives the introductions of CSCWR research and design activities in nuclear power institute of China. These activities include the CSCWR R&D group and current investments in NPIC, the concept design activities, the materials research activities, the thermal-hydraulic research activities, the irradiation test abilities and research activities, the water chemistry facilities and equipments and the potential collaborations. Those activities are moving on systemically and a wide domestic and international cooperation and collaboration is needed.

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SIMULATION OF A LARGE-BREAK LOSS OF COOLANT ACCIDENT IN THE HIGH PERFORMANCE LIGHT WATER REACTOR

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VTT Technical Research Centre of Finland took part in the recently ended European HPLWR2-project, in which a SCWR concept called the High Performance Light Water Reactor (HPLWR) was studied. VTT participated in the project by calculating preliminary safety analyses using two simulation tools, APROS and TRAB-3D/SMABRE, both of which were developed in-house. In order to support simulation of thermal hydraulics at supercritical pressures, these codes were modified by adding new constitutive equations suitable for the supercritical-pressures conditions, extending and refining the steam tables used to calculate the thermophysical properties of water, and by developing a numerical scheme which handles the liquid-to-supercritical-to-gas phase transition in a sound manner.

In this paper, an analysis of a large-break loss of coolant accident in the High Performance Light Water Reactor, calculated with APROS, is presented. The simulation is calculated with the one-dimensional separate two-fluid (6-equation) thermal-hydraulic flow model, and the neutronic behavior of the nuclear reactor is modeled using a three-dimensional neutron diffusion model. The simulation model, which represents the reactor concept with a three-pass core, includes the reactor pressure vessel, and a part of the steam line until the main steam line isolation valve. In the neutronic model, each fuel assembly cluster is modeled separately, but on the thermal hydraulic side only three core-flow channels, each corresponding to a single pass through the reactor core, are modeled due to computation time requirements.

The accident is initiated by a 1 x 100 % break in one of the four steam lines, between the pressure vessel outlet and the main steam line isolation valve. Decreasing pressure at the pressure vessel outlet initiates the reactor scram and closure of the isolation valves. Reactor inlet flow is kept constant until the feed-water tank runs empty. After the closure of the main steam line isolation valves, all the water flowing out of the pressure vessel runs through the break orifice into the containment. Low head safety injection, an active safety component, starts to inject cold water at the reactor inlet as the pressure has decreased sufficiently.

The presented analysis suggests that the reactor core of the HPLWR can be kept sufficiently cooled-down in the case of a large break LOCA in the main steam line using the specified safety systems. Some uncertainty to the simulation results is caused by the constitutive equations used at supercritical pressures, but they have very limited effect on the overall results in a simulation case, where the pressure drops to subcritical conditions at a very early stage. Also the current model with only three thermal hydraulic core channels is sufficient for this kind of simulation, where the reactor is brought to decay heat very early on to the simulation. Proper modeling of flows in each fuel cluster would be needed for more elaborate analyses, and for example analyses of reactivity initiated accidents.

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RELAP5/MOD3.3 AND TRACE 5.0 PREDICTIONS OF HEAT TRANSFER AND STABILITY FOR SUPERCRITICAL WATER FLOW IN HEATED PIPE

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This paper summarizes the activity performed for assessing the capabilities of RELAP5/Mod3.3 and TRACE5.0 codes at supercritical pressure conditions in heated channels. The study, conducted in the framework of the IAEA CRP "Heat Transfer Behaviour and Thermo-Hydraulics Code Testing for SCWRs"., includes two main activities. The former is related with the implementation into the sources codes of the correlation of Mokry et Al., developed for supercritical water in vertical channel. The achieved results have been compared with the experimental data from Task#9 of IAEA CRP benchmark.

The second activity aims to assess the behaviour of the two codes in the prediction of the stability conditions in heated channels with water at supercritical pressure in vertical and horizontal channel. The dimensionless numbers, introduced by Ambrosini [Ambrosini 2009], have been used to present the results and to identify the instability map regions.

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SOME PERTINENT ASPECTS OF COMENA R/D ACTIVITIES IN THE FIELD OF SCWR*

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Nuclear energy provides now over 16 percent of the world's total electricity and has the potential to contribute much more, especially if greenhouse concerns lead to a change in the relative economic advantage of nuclear electricity.
The recent technological advances in the field of nuclear power plant design together with the strong and continuous increase in the price of oil and gas are making the option of using nuclear energy for the production of electricity and water desalination a very attractive alternative to be included in the energy mix of many developing countries.
In Algeria, the strong rate of increase in the energy demand together with the presence of non negligible natural reserves of uranium in the Hoggar region and the availability of basic nuclear infrastructures capable of supporting reliably the introduction of nuclear power in the country, have prompted these past years governmental authorities to decide on the implementation of the nuclear power alternative in the country energy mix.
With respect to this point, extensive energy planning studies have been performed and indicated the need to put into operation a first nuclear power plant (capacity ≈ 1000 - 1200 MWe) by the year 2022 and a second plant of the same capacity by 2027. Furthermore, it is expected that the base load in the national electric generating capacity, for the period 2030-2050, will rely on nuclear.
To support such a program a nuclear technology investigation group was appointed within the Algerian Atomic Energy Commission (COMENA). The group recognized very quickly the importance of light water reactor (LWR) technologies for the Algerian nuclear power program.
Since the current evolutionary trend of LWR technologies is shifting toward the development of safer, simpler and more economical revolutionary concepts a special multidisciplinary team was set up to follow up and contribute as well as to the global R/D efforts in the domain of Super Critical Water Reactors (SCWR).
Pertinent aspects of ongoing activities at COMENA in the field of neutronics, thermal hydraulics and experimental parameters measurements are reviewed.

* To be presented at the TM on heat transfer, thermal-hydraulics and system design for supercritical pressure water cooled reactors, 5-8 July 2010 ; Pisa, Italy.

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ROADMAP FOR CONCEPTUAL DESIGN OF A SUPERCRITICAL PRESSURE WATER REACTOR IN SNERDI

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The supercritical water-cooled reactor (SCWR) was recommended as one of six promising Generation-IV nuclear reactor concepts by the GEN-IV International Forum (GIF).SCWR is the only reactor type with water as coolant. Due to its economical advantage, technology and experience continuity, SCWR has attracted considerable interests of nuclear industries and research institutions. In order to ensure the long-term nuclear power development in China, it is of crucial importance to employ the innovative nuclear systems of generation IV (GEN-IV).
SCWR is recognized as a natural extension of the existing commercial pressurized water reactors in China.This paper presents the main technical features of SCWR and its position in the Chinese long-term nuclear power development. A feasibility study on SCWR power plant in Shanghai Nuclear Engineering Research and Design Institute (SNERDI) has been initiated since this fiscal year. A R&D Roadmap of the supercritical pressure water reactor for the years 2010-2015 in SNERDI was proposed. It comprises feasibility studies for a thermal neutron spectrum core design, the reactor coolant system, the startup pressure- and volume- control system, the normal residual heat removal system, the active and passive safety system, and the balance of plant.
Basic technologies for supercritical pressure water reactor power plant like materials and code development, and the related heat-transfer investigations, critical flow experiments and material testing, are major R&D tasks. Finally, the ongoing research and development activities in SNERDI were summarized and the future needs were clarified.

Key Words
SCWR; Supercritical; Generation IV; Feasibility study; Roadmap