IAEA-TECDOC-1869

COMPANION CD-ROM

EDITORIAL NOTE

For further information on this publication, please contact:

Nuclear Power Technology Development Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
Email: Official.Mail@iaea.org

Enquiries should be addressed to the IAEA Publishing Section at:

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 26007 22529
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
www.iaea.org/books

STATUS OF RESEARCH AND TECHNOLOGY DEVELOPMENT FOR SUPERCRITICAL WATER COOLED REACTORS
IAEA, VIENNA, 2019
IAEA-TECDOC-1869 (COMPANION CD-ROM)
ISBN 978–92–0–101919–6
ISSN 1011–4289

Produced by the IAEA in Austria
April 2019

ANNEX (ATTACHED CD-ROM)

Status of Research and Technology Development for Supercritical Water Cooled Reactors

This Annex provides a collection of Abstracts and presentations from the second Technical Meeting on *Heat Transfer, Thermal Hydraulics and System Design for SCWRs* in Sheffield, UK, 22–24 August 2016; and the third Technical Meeting on *Materials and Chemistry for SCWRs* in Řež, Czech Republic, 10–14 October 2016.

Presentations from the second Technical Meeting on *Heat Transfer, Thermal Hydraulics and System Design for SCWRs* are numbered 1–13.

Presentations from the third Technical Meeting on *Materials and Chemistry for SCWRs* are numbered 14–25.

Presentation File Name: 1_TM52353_TS-I-1_Leung.pdf

THERMAL-HYDRAULICS STUDIES OF THE CANADIAN SCWR PROGRAM

Laurence K.H. Leung
Chalk River Laboratories
Canadian Nuclear Laboratories
Chalk River, Ontario K0J 1J0
CANADA

Canada is developing a pressure-tube-type Super-Critical Water-cooled Reactor (SCWR) concept with enhanced safety, proliferation resistance and sustainability characteristics at compatible costs to current fleet of nuclear reactors. It joined the Generation-IV International Forum for cooperative research and development (R&D) with the international community and is currently collaborating with researchers in China, European Union, Japan and the Russian Federation in developing SCWR concepts. Despite of differences in the core configuration of SCWR concepts being pursued in various countries, technical issues encountered in several technology areas are common and can be addressed collectively.

Thermal-hydraulics at supercritical pressures has been identified as one of the critical technology areas in the development of the SCWR concept. It affects the operating power and the safety margin of the SCWR concept, and also has an impact on the selection of cladding material and neutronic design. Canada has initiated a national program in support of R&D for the SCWR. Thermal-hydraulics studies have been performed at Canadian Nuclear Laboratories and Canadian universities. A number of joint projects were established to extend the thermal-hydraulics R&D effort through bilateral collaborations between Canadian Nuclear Laboratories and researchers in China, the Russian Federation and the United Kingdom. In addition, Canada's participation in the Cooperative Research Project hosted by the International Atomic Energy Agency has facilitated further enhancement of the experimental database and the analytical tools.

A summary of the thermal-hydraulics studies in the Canadian SCWR program is presented. Heat-transfer experiments performed with tubes, annuli and bundles in water, carbon dioxide or refrigerant flows are described. Experimental data obtained from these experiments have been examined for investigating separate effects on heat transfer (such as diameters, spacing devices, etc.). Analytical tools have been assessed against the latest experimental data for quantifying the prediction capability and applicability. These assessment results are presented.

Presentation File Name: 2_TM52353_TS-I-2_Starflinger.pdf

SCWR – REVIEW OF THE RESULTS OF THE PROJECT "HPLWR PHASE 2"

Joerg Starflinger,
University of Stuttgart, Institute of Nuclear Technology and Energy Systems (IKE),
Pfaffenwaldring 31, D-70563 Stuttgart, Germany

Thomas Schulenberg,
Karlsruhe Institute of Technology, Institute for Nuclear and Energy Technologies (IKET),
Hermann-von Helmholtz Platz 1, D-76344 Eggenstein-Leopoldshafen

Core design of supercritical water cooled Reactors (SCWR) is strongly influenced by the variation of thermodynamic properties of supercritical water near the critical point. Operating at a design pressure of 25MPa with a heat up from 280°C to 500°C, for instance, changes the density of supercritical water smoothly and continuously from a liquid-like state to a gas-like state. However, this smooth and continuous transition is non-linear. From the thermohydraulic design point of view, this non-linearity provides a special challenge in a hot-channel analysis. By definition, a hot-channel is a sub-channel somewhere inside the core, where all peaking factors (core-wise, assembly-wise), all uncertainties (experimental, numerical) and allowances (for plant operation) are condensed into one channel. The hot channel factor is multiplied to the nominal heat-up within the average channel to gain the enthalpy rise in the hot channel. The properties in this hot channel may not exceed the operational design requirements.

The hot channel method is a conservative method, because all uncertainties and peaking factors are applied to one channel, only. From the design point of view of a nuclear reactor not being built up to now, conservative methods should be applied at the first stage providing a suitable safety margin right from the beginning of the design.

Applying the hot channel factor analysis to the expected heat-up of a SCWR resulted in the three-pass core design as analysed in the HPLWR Phase 2 project. The three passes resulted from necessity to mix the supercritical fluid during the heat-up inside the core to avoid hot streaks, which could challenge the cladding material. For the HPLWR project, a maximum cladding surface temperature of 630°C was selected. Three pass simply means that the flow passes the reactor core three times: upward flow in the core – mixing in an upper plenum - downward flow in a different section of the core – mixing in a lower plenum – upward flow in a different section and delivery of supercritical water of 500°C to the turbine.

Several options can be applied to reduce the number of passes through the core. All of them have in common that either the hot-channel factor will have to be reduced, or the operation limits will have to be increased.

The first option is to select a material of the cladding which can withstand higher maximum cladding surface temperatures (increasing the operational limit). The higher the temperature, the more the material tends to increased corrosion, stress-corrosion cracking, or creep. High chromium content or nickel-base alloys are a possible path to go forward; however, low mechanical strength and increased parasitic neutron absorption must be taken into account.

Another option is to mix the supercritical fluid already in the fuel assembly. In the HPLWR Phase 2 project, wire wrap spacers (known from the breeder design) were foreseen instead of grid spacers which are used in BWR and PWR. Such wire wrap spacers mix quite well, but could possibly lead to hot spots on the cladding surface, which must be avoided though a suitable design.

Although the flow is well mixed within an assembly, the non-uniform neutron flux (and power) profile across an assembly still results in a non-uniform heat-up of the supercritical fluid. In the HPLWR design, three smaller assemblies were grouped into one assembly cluster with common head- and foot piece. The grouping was selected because control rod devices from existing PWRs (incl. control rod drives) should be used and the allowed number of flanges in the closure head of the reactor was limited at the increased pressure. However, for flattening of the neutron flux and power profile, smaller fuel assemblies (like in BWRs) would be more appropriate, which are also more flexible for shuffling.

The last option would be to reduce the mean core outlet temperature. In this case, the hot channel does not exceed 500°C whereas the nominal channel will be heated up to 390°C, only. Using this option, the supercritical water cannot be delivered directly to the turbine because of the liquid-like state. In this case, the SCWR design would be close to a PWR design with a closed primary system and with steam generators.

These options should be further investigated for the deployment of a Supercritical Water Cooled Reactor.

Presentation File Name: 3_TM52353_TS-I-3_Kuzmichev.pdf

HEAT TRANSFER PROBLEMS IN FA OF VVER-SCP

A.N. Churkin, P.V. Yagov, A.Y. Kuzmichev, Y.O. Yagova
OKB “GIDROPRESS”, Podolsk, Russia

Three concepts of reactor plant (RP) with supercritical water coolant VVER-SCP (SCWR), which refer to the Generation-4 systems, were developed in Russia:

— VVER-SCP-1700 - one-circuit high-power RP, with fast-resonant neutron spectrum in the core;
— SPCS-600 - two-circuit RP, with pseudo-vapor supercritical primary coolant and fast neutron spectrum in the core;
— VVER-SCP-I - two-circuit RP, with supercritical water coolant in the primary circuit and regulated neutron spectrum.

Up to the moment, OKB "GIDROPRESS" implements engineering development of VVER-SCP-1700 and performs thermal-hydraulic analyses.

Several problems, which were discovered during the subchannel calculations of the coolant flow in FA of one-circuit VVER-SCP-1700 must be studied with the first priority.

The report emphasizes the problem of the displacement of coolant from the more heat-stressed part of the bundle to the less heat-stressed. The results of the calculations concerning this problem are presented. New specification of the future thermal-hydraulic subchannel calculations in the framework of development and verification of TEMPA-SC code is presented.

Presentation File Name: 4_TM52353_TS-II-1_Kline.pdf

EFFECT OF TUBE DIAMETER ON SUPERCRITICAL HEAT TRANSFER TO CO_2

N. Kline, S. Tavoularis
Department of Mechanical Engineering, University of Ottawa, Ottawa, ON, Canada

Extensive convective heat transfer measurements have been collected in the Supercritical University of Ottawa Loop (SCUOL) with carbon dioxide at supercritical pressures flowing vertically upwards in tubular test sections having inner diameters equal to 22.0, 8.0, and 4.6 mm. Outer wall temperature was measured by a large number of thermocouples, from which the inner wall temperature and the local heat transfer coefficient were estimated. The measurements extend over wide ranges of conditions, which cover both the normal and deteriorated heat transfer modes.

Experiments were conducted using all three test sections with the purpose of determining the tube diameter effects on the local heat transfer coefficient h for normal heat transfer. Measurements were collected for 15 different flow and heating conditions, all at $P/P_{critical} \approx 1.13$. The results showed that, in most cases, the corresponding h was higher for smaller diameter tubes. Nevertheless, for flows at low mass fluxes ($G \leq 400$ kg/m^{2}s), the h values from the 8 and 22 mm test sections were comparable, especially at high specific bulk enthalpies ($H_b > 300$ kJ/kg); for $G = 300$ kg/m^{2}s, the h values from all three test sections in the high enthalpy range actually nearly coincided. At higher mass fluxes, however, the h values remained distinct for each test section over the full range of H_b values examined, with higher h values for smaller test section diameters.

Figure 1 summarises some of the results. To quantify empirically the diameter effect, we have used the diameter $D_{ref} = 8$ mm as a reference and normalized h by the corresponding value h_{ref} in the 8 mm test section under the same conditions. To select a suitable function form for fitting, we assumed that h/h_{ref} should approach an asymptote from above, as $D/D_{ref} \to \infty$, which corresponds to a plane channel. It was found that h/h_{ref} could be fitted roughly by an exponential function containing a single coefficient A, which could be expressed as a function of G/G_{ref}, where $G_{ref} = 1380$ kg/m^{2}s, as follows:

$$h/h_{ref} = \frac{A}{1-(1-A)^{D/D_{ref}}}, \qquad A = -\left(G/G_{ref}\right)^2 + 0.1\left(G/G_{ref}\right) + 1$$

This fit is only intended to provide approximate scaling estimates within the range 300 kg/m^{2}s $\leq G \leq$ 700 kg/m^{2}s. Each curve in Fig. 1 represents an average estimate, which masks the fact that the above function does not take into account the effect of H_b, which is significant at higher G.

To determine the conditions at the onset of heat transfer deterioration (HTD), experiments were also performed by gradually increasing the wall heat flux q, while keeping the pressure constant at $P \approx 1.13P_{critical}$, the mass flux constant in the range 200 kg/m^{2}s $\leq G \leq$ 1000 kg/m^{2}s and the inlet temperature T_{in} constant in the range 0 ºC $\leq T_{in} \leq$ 35 ºC, until a temperature spike was observed in the wall temperature profile. It was found that, at the onset of HTD, the heat flux limit q^* could be fitted

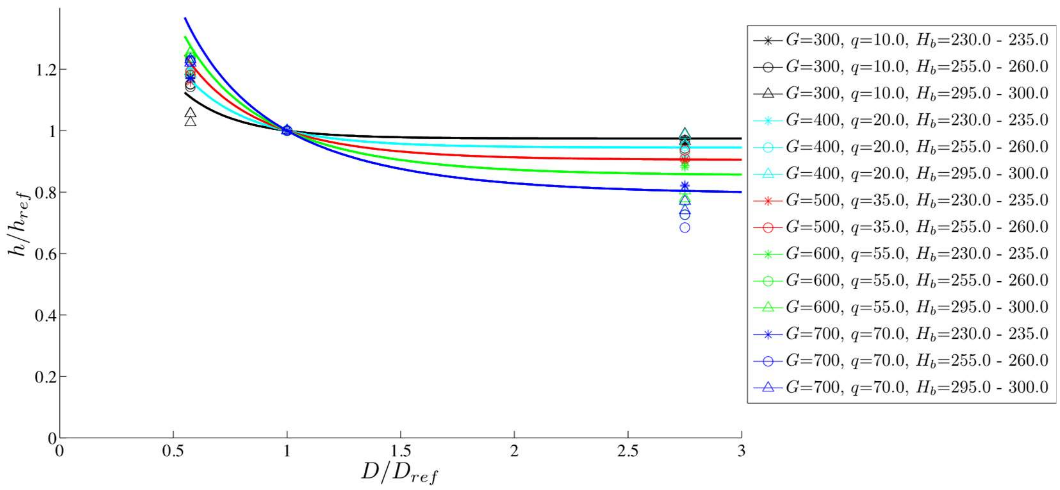

Figure 1: Normalized heat transfer coefficient, using the 8 mm results as a reference, versus the normalized diameter; a family of lines was fitted to data for different measured mass fluxes. The legend gives values of mass flux G in kg/m^{2}s, heat flux q in kW/m^2, and specific bulk enthalpy H_b in kJ/kg.

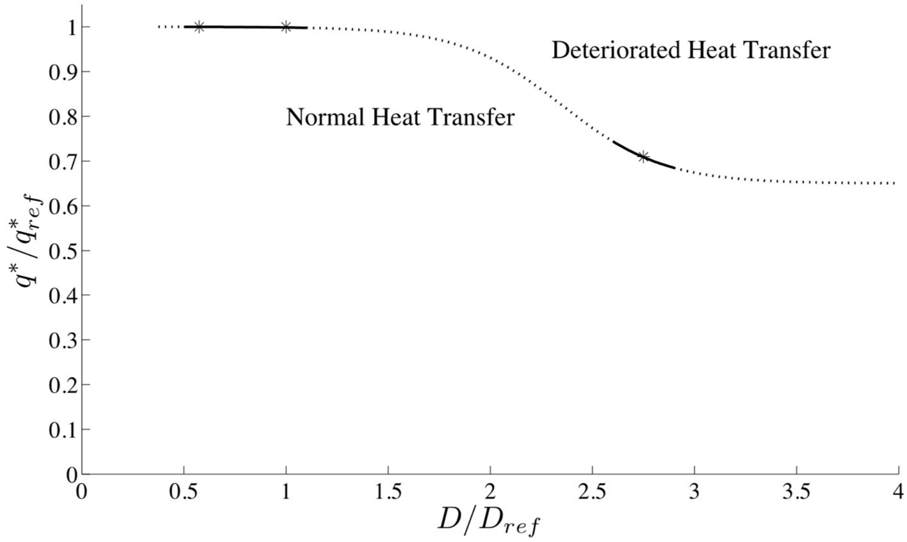

Figure 2: The normalized heat flux limit versus the normalized diameter, valid for G = 300 to 1000 kg/m^{2}s; symbols represent measured heat flux limits; solid and dotted lines represent a fitted function. by a power law of *G* with the same exponent for all three test sections. The heat flux limit for the 8 mm test section, used as a reference, could be represented as $q^*_{ref} = 6.2 \times 10^{-5} G^{2.2}$. The proportionality constant for the 4 mm tube was essentially the same, whereas for the 22 mm tube it was significantly lower (Fig. 2).

Presentation File Name: 5_TM52353_TS-II-2_Dubey.pdf

DEVELOPMENT OF CORRELATION FOR HEAT TRANSFER ENHANCEMENT AND DETERIORATION FOR SUPERCRITICAL FLUID USING FREON 22 EXPERIMENTAL RESULTS

S.K.Dubey[1], R.P. Vedula[2], Kannan N. Iyer[2] and Avinash J. Gaikwad[1]

1 NSAD, Atomic Energy Regulatory Board, Anushaktinagar, Mumbai, India 400094

2 Dept. of Mech. Engg., IIT Bombay, Powai, Mumbai, India-400076

The heat transfer behaviour of a supercritical fluid is an important input for the performance evaluation of SuperCritical Water-cooled Reactors (SCWR) of Generation IV nuclear power plants. SCWRs have very high overall thermal efficiency of about 45-50% because their operating pressures and temperatures are high. Accident analyses for licensing are carried out using system thermal hydraulics codes in which well established and validated correlations for Heat Transfer Coefficient (HTC) have to be built in. Experimental studies have shown that there is Heat Transfer Enhancement (HTE) for supercritical fluid near the pseudocritical temperature at relatively low heat flux to mass flux ratios. The peak HTC decreases as this ratio increases. At very high values of heat flux, a peak in wall temperature appears due to Heat Transfer Deterioration (HTD). This phenomenon is important as the nuclear fuel clad may fail at high temperatures induced by HTD and result in release of radioactive nuclides into the coolant streams. It is also important for the sizing of core. Several investigators have carried out experiments using water, CO_2 and R22 etc. due to similarity in their thermophysical properties, and it was observed that the bulk fluid enthalpy at which peak wall temperature appears is different at different test specifications (heat flux, mass flux and different inlet temperature). Effects of inlet temperature are found to be significant for HTD but results reported in literature have not reported this effect in detail. Experimental wall temperatures available in open literature were measured by thermocouples positioned at regular intervals. However, since the HTD is shown to result in very steep temperature changes, highly local temperature measurements are desirable. HTC correlations are available which are able to predict HTE satisfactorily but HTD predictions from available correlations are poor and therefore, better correlations are required to predict HTD.

An experimental facility has been built with R22 as working fluid which acts as the simulant fluid for water. The appropriate similarity and scaling between thermophysical properties, process conditions and heat transfer behaviour were established. R22 was used as the working fluid since data for water at protypical conditions can be obtained at low operating pressure and temperature resulting in reduced power requirement and increased safety. A Supercritical Freon Test Facility was then designed and built with two vertical tubular test sections of ID equal to 6 mm and 13.5 mm respectively. Experiments with vertically upward flow at 55 bar system pressure were carried out and a thermal camera was used to obtain wall temperatures every 1-1.5 mm along the axial direction. Experiments were conducted with water before using R22 to validate the experimental and data reduction procedures. Initial experiments with R22 were conducted to demonstrate the reduction in peak HTE with increase in heat flux. As low heat flux is needed to carryout experiments for HTE, the rise in bulk fluid temperature across the test section will be very low especially near the pseudocritical temperature and therefore, experiments were carried out in a piece-wise fashion. Experiments were performed by changing the inlet temperature with the help of a preheater and using a sufficient overlap between experiments, the results were joined together. The results showed that the HTE is independent of inlet temperature and the data could be properly integrated together to obtain the behaviour over the required length. Experiments were then conducted for HTD at several heat and mass flux values and inlet temperatures. It was observed from experimental results that the onset of HTD occurs when q/G is more than 0.056 to 0.072 kJ/kg and when the inlet temperature was lowered, the onset of HTD appeared at a relatively higher q/G. The bulk fluid enthalpy and temperature at which onset of HTD appeared also reduced when the inlet temperature was decreased. A sharp rise in the wall temperature was initiated when the wall temperature was just above the pseudocritical temperature at relatively high heat flux values. Two peaks in wall temperature were observed in the results at the lower inlet temperature. It was observed that that bulk fluid enthalpy at which peak HTD occurred decreases with increase in heat flux. The heat transfer coefficient values in the region of the test section before the start of HTD were in good agreement with the predictions using the Dittus-Boelter correlation and an appreciable recovery of heat transfer coefficient after the deterioration was observed at relatively low heat fluxes in the HTD zone. Typical experimental wall temperatures at various heat flux values are shown in Fig. 1 and effect of inlet temperature on the onset of HTD is shown in Fig. 2. Each curve shown in these figures includes about 1500 axial points for wall temperature and while the nature of the curve is obtained by using all the points, only a few are shown for legend on the figure for increased clarity.

A new non-dimensional number $ND_0 = \frac{q\mu_{pc}}{\mathrm{G}\mathrm{Pr}_{pc}^{0.5}k_{pc}\Delta\mathrm{T}_0}$ has been identified and the criteria for onset of heat transfer deterioration is proposed to occur when $ND_0 > 2.18\times10^{-4}$. Separate correlations to predict heat transfer coefficient for heat transfer enhancement and deterioration have been developed based on the present experimental results, for example, HTC correlation given by Eq. (1) is developed for reduction in heat transfer enhancement. The correlations are based on non-dimensional numbers and therefore can be applied to any supercritical fluid. These correlations were used for reported data employing water and CO_2, and the comparisons were noticed to be satisfactory.

$$Nu_b = 0.023\,\mathrm{Re}_b^{0.8}\mathrm{Pr}_b^{0.4}\left(\frac{C_{pref}}{C_{pb}}\right)^x, x = 1868ND_0 - 0.103 \tag{1}$$

where C_{pref} = 4.112 kJ/kg K for water and similar values can be obtained for other supercritical fluids using the appropriate property values.

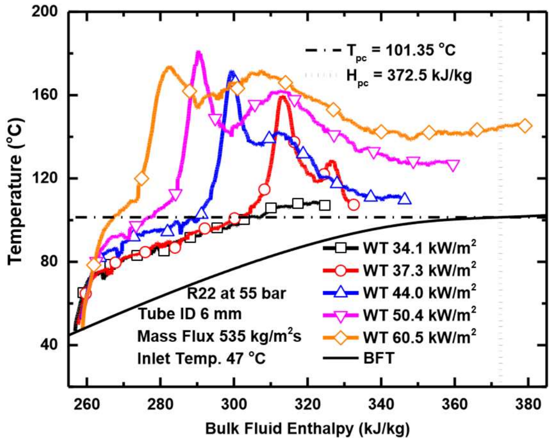

Fig. 1 Experimental wall temperature along the bulk fluid enthalpy

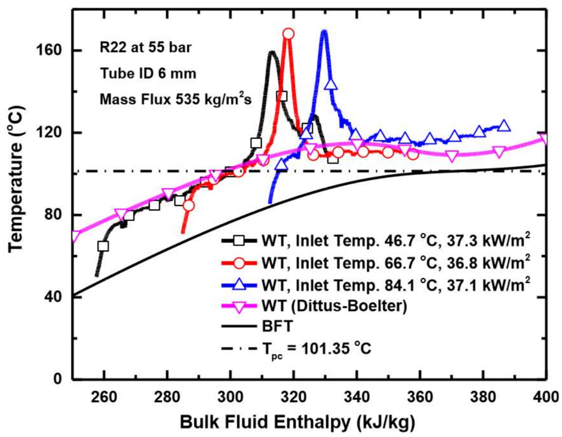

Fig. 2 Effect of inlet temperature on wall temperature at onset of HTD

Keywords: Generation IV Reactor, Supercritical water Cooled Reactor (SCWR), Heat Transfer Deterioration (HTD), Heat Transfer Enhancement (HTE), Supercritical Freon Test Facility (SFTF)

Presentation File Name: 6_TM52353_TS-II-3_Pucciarelli.pdf

FLUID TO FLUID SCALING OF HEAT TRANSFER WITH SUPERCRITICALPRESSURE FLUIDS: RECENT CONSIDERATIONS AND FUTURE PERSPECTIVES

Andrea Pucciarelli[a], Walter Ambrosini[a], Shuisheng He[b] and Wei Wang[b]
[a] Dipartimento di Ingegneria Civile e Industriale, University of Pisa
[b] Department of Mechanical Engineering, University of Sheffield

The contribution reports on the ongoing work at the University of Pisa about fluid to fluid scaling with supercritical pressure fluids. The first studies in this regard started in 2006 when relevant non dimensional quantities were introduced for analysing flow instabilities, supposing the existence of similarities between the exceeding of the pseudo-critical temperature in supercritical fluids and the boiling threshold in subcritical ones. Since the beginning interesting features where noticed, e.g. the fact that the non-dimensional density trend (ρ^*) seems to be a function of the dimensionless enthalpy (h^*) only, regardless of the supercritical pressure and, in general, of the fluid.

This feature allowed the research team in Pisa to successfully find out similarities between the dynamic behaviour of different supercritical fluids in forced and natural convection. In fact, as the $\rho^* - h^*$ trend is similar for different fluids, once the same h^*_{in} and the same Δh^* along the same heated length (called N_{TPC}) are imposed, the fluids should undergo similar buoyancy effects. In particular, the same h^*_{bulk} and ρ^*_{bulk} are expected in correspondence of the same axial position.

While obtaining success for dynamic behaviour, mainly driven by bulk fluid effects, difficulties were found when facing heat transfer phenomena. In fact, incompletely coherent behaviours were noticed when comparing different fluids adopting the recipe that proved to be suitable for studying instabilities. In particular, Carbon Dioxide tended to return stronger heat transfer deterioration phenomena than the ones reported by Water once “similar” boundary conditions were set.

The main reason of this discrepancy was due to the fact that, though for global flow dynamics by imposing Δh^* is sufficient for defining the density differences which generate the leading forces of the phenomenon, in forced convection the local value of heat flux becomes very important, since different fluids show different heat transfer capabilities. In particular, this happens because, unlike density, there is no close similarity between the dimensionless trends of specific heat, thermal conductivity and dynamic viscosity when considering different fluids. In the frame of the latest works the problem was solved, at least basing on CFD data, with the help of RANS calculations. Instead of imposing the same Δh^* along the same heated length, a different length is defined scaled through the values of Stanton number. Though the same N_{TPC} is maintained, no geometrical similarity in the L/D ratio exists anymore and now the same h^*_{wall} value is expected when the same h^*_{bulk} conditions are obtained. This technique was tested over a small number of selected cases adopting four different supercritical fluids. The analyses returned promising results; very similar h^*_{wall} trends were obtained and similarities were also noticed for the dimensionless velocity, enthalpy and turbulent kinetic energy radial trends.

Unfortunately, no experimental evidence exists at the moment for the proposed scaling method. This is mainly due to the fact that experiments adopting Carbon Dioxide as working fluid often consider very near critical conditions, corresponding to bulk values up to 350 °C for water, which certainly deserve particular and expensive facilities for being carried on. Consequently, new ad hoc experimental campaigns should be planned for proving whether or not the proposed technique can be relied upon.

With the aim of obtaining more realistic results, analyses are currently being performed at the University of Pisa by adopting LES techniques and are summarised in the present paper. Carbon Dioxide is considered for the reference case and scaling attempts adopting water are currently running. The proposed results are not definitive yet; however, some interesting features can be observed: the calculations return quite similar trends though, due to the low Reynolds number of the selected case, fully developed turbulent flow is not assured.

A parallel work, carried on in cooperation with the University of Sheffield, takes into account DNS techniques for performing the scaling attempts. RANS calculations were performed in order to plan the DNS campaign by selecting suitable boundary conditions. Three scaling cases were initially planned, nevertheless, at the moment, only one seems having reached sufficiently stationary conditions. The results obtained in the mentioned case are here presented and commented.

Many analyses are then still needed, nevertheless the obtained results seem to be promising. The main goal of the present contribution is then trying and show the effectiveness of the proposed scaling technique and, moreover, to persuade researchers to try and consider the proposed method as a cue for performing experimental campaigns.

Presentation File Name: 7_TM52353_TS-II-4_Kim.pdf

ROLE OF NEAR-WALL TURBULENT STRUCTURES IN THE HEAT TRANSFER OF TRANSCRITICAL CHANNEL FLOW

Kukjin Kim[1], Jean-Pierre Hickey[2] and Carlo Scalo[1]

[1] School of Mechanical Engineering, Purdue University, 585 Purdue Mall, West Lafayette, IN 47907-2088, USA

[2] Department of Mechanical and Mechatronics Engineering, University of Waterloo, 200 University Avenue W., Waterloo, Ontario, N2L 3G1, Canada

We report on a series of direct numerical simulations of a canonical channel flow setup to characterize the turbulent heat transfer effects in a transcritical thermodynamic regime for tetrafluoroethane (R-134a). To unambiguously quantify the turbulent heat transfer, the top and bottom channel walls are maintained at different isothermal temperatures based on a selected temperature difference, ΔT, above (top wall) and below (bottom wall) the pseudo-boiling (PB) of the fluid (determined based on the bulk pressure of the channel). The physical setup and thermodynamic states are shown in figure 1. This setup allows us to study the thermodynamic region in which the fluid undergoes the greatest thermophysical variation in a statistically-steady state. The channel flow is simulated using a high-order compact finite difference scheme (6th-order accuracy) for the solution of the conservative Navier-Stokes equation. The equations are closed with a cubic Peng-Robinson equation of state and Chung's model is used to account for the thermophysical properties of the fluid. A rigorous grid convergence study is undertaken to show that characteristic point-to-point oscillations emerging from the flux calculations (for conservative schemes) can be bounded with sufficient spatial resolution. Therefore, a high-order simulation, which fully resolves all scales of turbulent motion, can be attained without additional artificial dissipation, limiters or non-conservative formulations.

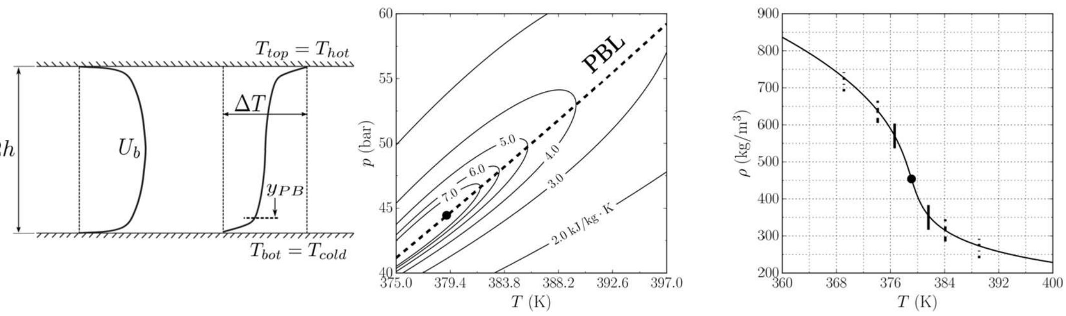

Figure 1: Physical setup and characteristic dimensions of the investigated channel flow (left). Pressure - temperature phase diagram showing the pseudo-boiling line (PBL) and the isolines of specific heat; the circle corresponds to the critical point (middle). Profile of density with temperature at p=1.1 p_c with the investigated conditions shows as vertical lines for ΔT=5, 10 and 20 K (right).

Turbulence is the primary contributor to the near wall heat transfer and its effect is modulated by the underlying non-linear thermodynamics. Time-averaged density and temperature profiles (see figure 2) show the highest thermodynamic gradients are located near the walls with a qualitatively symmetric

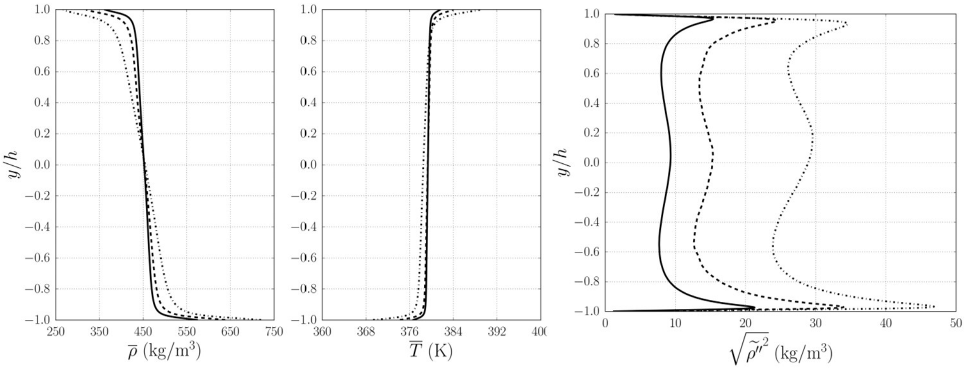

Figure 2: Mean density (left) and temperature (middle) profile across the channel and the rms of the density fluctuations (right). The lines correspond to ΔT = 5 K (——), 10 K (- - -), and 20 K (-·-).

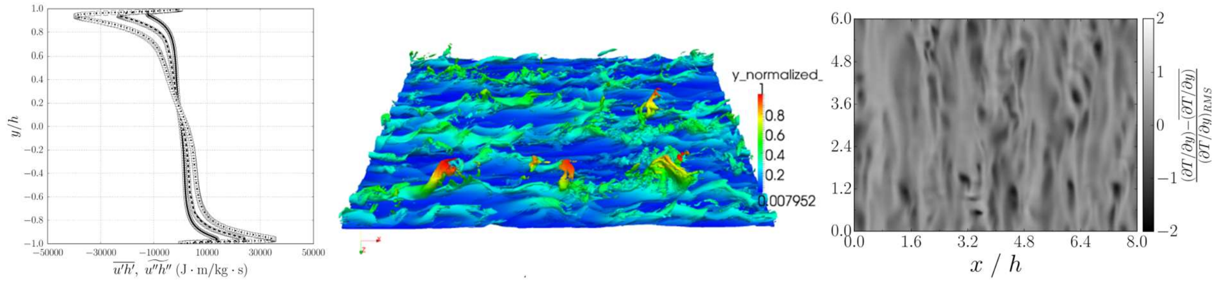

Figure 3: Turbulent enthalpy flux using the Reynolds- and Favre-averaged fluctuation quantities at ΔT = 5 K (#; ——), 10 K (4; - - -), and 20 K (2; -·-) where symbols and lines are for the Reynolds- and Favre-averaged fluctuation quantities, respectively. Turbulent structures based on the isosurfaces of the Q-criteria colored by the distance from the wall (middle) at ΔT=20K. Instantaneous normalized temperature gradients at the bottom (right) wall.

distribution about the centerplane. The root mean square (rms) of the density has typical peaks in the near wall region. As the temperature difference between the top and bottom wall increases, so does the asymmetry of the density fluctuations. Enhanced fluctuations are observed near the cold wall with increasing ΔT, but more importantly, we note the emergence of a distinct centerplane peak, see figure 2 (right). A similar anomalous behaviour is observed in the total enthalpy flux in the streamwise direction in figure 3 (left). Here we note a discontinuous profile of the enthalpy flux near the centerline when ΔT increases. Near wall turbulent structure visualizations reveal that the near wall heat and momentum transfer is primarily governed by the ejection of dense fluid lumps away from the cold wall (bottom) towards the center of the channel. This ejection process occurs as streamwise aligned vortical structures lift-up the high density fluid (below the PB point), and given its increased momentum, these lumps migrate to the center of the domain where the elevated temperature results in a pseudo-transition away from the walls (see figure 3, middle). This mechanism leaves its mark on the instantaneous temperature gradients at the wall shown in figure 3 (right). This structural understanding in transcritical flow provides new insight into the mechanisms of heat transfer in these complex thermodynamic states. This work will be continued with the objective of developing relevant wall models for the accurate modelling of turbulent heat transfer in transcritical flows.

Presentation File Name: 8_TM52353_TS-II-5_Razumovskiy.pdf

THERMAL ACCELERATION OF SCW FLOW IN HEAT-GENERATING CHANNELS AS A FACTOR OF HEAT TRANSFER DETERIORATION

V.G. Razumovskiy, E.M. Mayevskiy, E.N. Pis'mennyi, A.E. Koloskov
National Technical University of Ukraine 'Kyiv Polytechnic Institute'

Numerous experimental studies of heat transfer and pressure drop in smooth tubes cooled by water at supercritical pressure (SCW) revealed that at certain high heat flux rates *q/G* and coolant temperature range from about 340 to 400 ºC, i.e., in the region of transition from "liquid" to "gaseous" state, when water viscosity is growing with temperature, hydraulic resistance to thermal acceleration of the flow Δp_{ac} could exceeds friction resistance Δp_{fr} (case of high acceleration condition). At the temperature below and above this range total pressure drop Δp is a function either of a very strong dependence of coolant's viscosity upon temperature (viscous flow, if $t_f < 150$ ºC) or of viscous and inertial forces, if $150 \leq t_f \leq 340$ ºC or $t_f \geq 400$ ºC.

The mentioned conditions of SCW flow high thermal acceleration correspond to two-dimensional thermal flow acceleration in both longitudinal and radial directions that could cause a significant error of acceleration pressure drop prediction by its routine dependence upon one-dimensional change in specific volume of coolant along heated section (for example, such calculation could result in frictional resistance factor reduced to isothermal one $\xi_{fr}/\xi_o > 1$).

To get correct values of acceleration resistance factor ξ_{ac} the IVTAN's method of two pressure drops tested at supercritical carbon dioxide for the first time was applied for SCW. All the experiments by the noted method conducted at the flow enthalpy within the range from 1370 to 2180 kJ/kg related to the mentioned region of transition from "liquid" to "gaseous" state.

Analysis of the obtained experimental data revealed that $\xi_{ac}/\xi_{fr} > 1$ under negligible effect of buoyancy is the necessary condition of heat transfer deterioration caused by laminarization of boundary layer and resulted in suppression of turbulence. Thus, these data could serve as an additional evidence of crucial effect of thermal acceleration on heat transfer and its deterioration under the conditions of the developed turbulent flow even without impact of free convection. The empirical correlation for prediction of boundary heat flux rate $(q/G)_b$ corresponding to transition from normal to deteriorated heat transfer based on 120 experimental tests that covered a wide range of q/G (from 0.3 to 1.5 kJ/kg) is proposed.

This correlation is in a good agreement (mainly within ± 25%) with the results obtained for 1-rod (annular channel), 3-rod and (tests are still in progress) 7-rod bundle imitators (about 200 experimental values of $(q/G)_b$).

Unlike the existing recommendations (by Petukhov, Yamagata, Gabarayev, Vikhrev et al.) it takes into account: (a) an initial thermal state of coolant in relation to it at the point of maximal isobaric specific heat capacity; and (b) the length of heated section from inlet to the point of DHT beginning. As a result, in many cases the DHT boundary could be reasonably almost twice as high as considered so far, thus substantially increasing safe thermal load of a channel.

It is found that even an insignificant instability in operating parameters, when $(q/G) > (q/G)_b$, is able to cause quick and wide variations (exceeding 50-100 ºC) in wall temperature. They negatively impact on reliability of heated surface despite usually in this case t_w is by 100…200 ºC below the temperature permissible by long-term strength conditions, i.e., is far from the level of extreme temperatures but in some works resulted in wall burnout.

Key words: method of two pressure drops, friction resistance, adiabatic pressure drop, flow laminarization, boundary heat flux rate

Presentation File Name: 9_TM52353_TS-III-1_Schulenberg.pdf

SIMULATION OF TRANSIENT HEAT TRANSFER IN AN SCWR FUEL ASSEMBLY TEST AT NEAR-CRITICAL PRESSURE

Thomas Schulenberg, Ivan A. Sierraalta Casas,
Karlsruhe Institute of Technology, Institute for Nuclear and Energy Technologies (IKET),
Hermann-von Helmholtz Platz 1, D-76344 Eggenstein-Leopoldshafen

Hongbo Li
China Nuclear Power Technology Research Institute Co., Ltd, Shenzhen, China,

While supercritical water is a perfect coolant with excellent heat transfer, a temporary decrease of the system pressure to sub-critical conditions, either during intended transients or by accident, can easily cause a boiling crisis with significantly higher cladding temperatures of the fuel assemblies. Such situation is planned to be tested with a small fuel assembly of 4 rods, to be operated in a critical arrangement with supercritical water inside a research reactor in the Czech Republic. First out-of-pile tests of this experiment have recently been performed in the SWAMUP facility at SJTU in China.

Some of the transient tests have now been simulated at KIT with a one-dimensional MATLAB code, assuming quasi-steady state flow conditions, but time dependent temperatures in the fuel rods. Heat transfer at supercritical and at near-critical conditions was modelled with a recent look-up table of Zahlan (2015), and sub-critical film boiling was modelled with the look-up table of Groenveld et al. (2003), giving the best accuracy at minimum run time of the code. Moreover, a conduction controlled rewetting process was included in the analyses, which is based on an analytical solution of Schulenberg and Raqué (2014). This fast method should be applicable later to any system code for nuclear application.

The new method could well reproduce the boiling crisis during depressurization from supercritical to subcritical pressure, including rewetting of the hot zone within some minutes, as shown in Fig. 1, but the peak temperature was somewhat under-predicted. Tests with a lower heat flux, which did not cause such phenomena, could be predicted as well. In another test with increasing pressure, however, a boiling crisis was also observed at a heat flux, which was significantly lower than the critical heat flux predicted by the CHF look-up table of Groeneveld et al. (2006).

The presentation is summarizing the physical models and the numerical approach. Comparison with experimental data is used to discuss the applicability of these look-up tables for the design of supercritical water-cooled reactors.

Presentation File Name: 10_TM52353_TS-III-2_Lyons.pdf

CFD BENCHMARK ANALYSIS FOR 2x2 FUEL ROD BUNDLE

Kate Lyons, Mark Anderson
University of Wisconsin – Madison (USA)

In order to better understand and adequately predict heat transfer in supercritical fluids, it is important to complete simulation benchmarks to compare the model results with experimental data. This benchmark process allows for the simulation model to be tuned to the fluid and geometry application to enable better future prediction capability of the model. As part of the IAEA Coordinated Research Project (CRP) on SCWR thermal-hydraulics, CRP participants validate their computational fluid dynamics (CFD) or subchannel code with four-rod bundle heat transfer experimental data. Data is being provided by both Shanghai Jiao Tong University (SJTU) and the University of Wisconsin-Madison (UW-Madison). In this study, progress towards CFD model validation against SJTU experimental data is presented.

In the test section of the UW-Madison facility flow loop, water flows upwards over four heated rods which simulate a two by two fuel rod bundle with a pitch to diameter ratio of 1.33. The rods are manufactured to create a cosine axial heat flux profile to better simulate reactor-like conditions. Thermocouples were swaged under the cladding of the heater rods at various axial and radial positions to provide temperature data for the system. The UW-Madison test section was simulated using the CFD program Fluent under subcooled water conditions to verify geometry and to establish an initial working model. These initial simulations were completed before extending the simulation domain to supercritical water and modelling the SJTU test section.

Many experiments have been completed using the SWAMUP test facility flow loop at SJTU to study heat transfer characteristics in supercritical and subcooled water. In the SJTU test section, four tubes are uniformly heated within the square flow channel with a pitch to diameter ratio of 1.18. There are six equally spaced grid spacers that hold these tubes in place. Each heated tube contains a sliding thermocouple that allows for temperature measurement along the axial length and at defined radial positions. A comparison of the UW-Madison and SJTU flow channel cross sections can be seen in Figure 1. The heat flux profile for each facility test section is shown here as well.

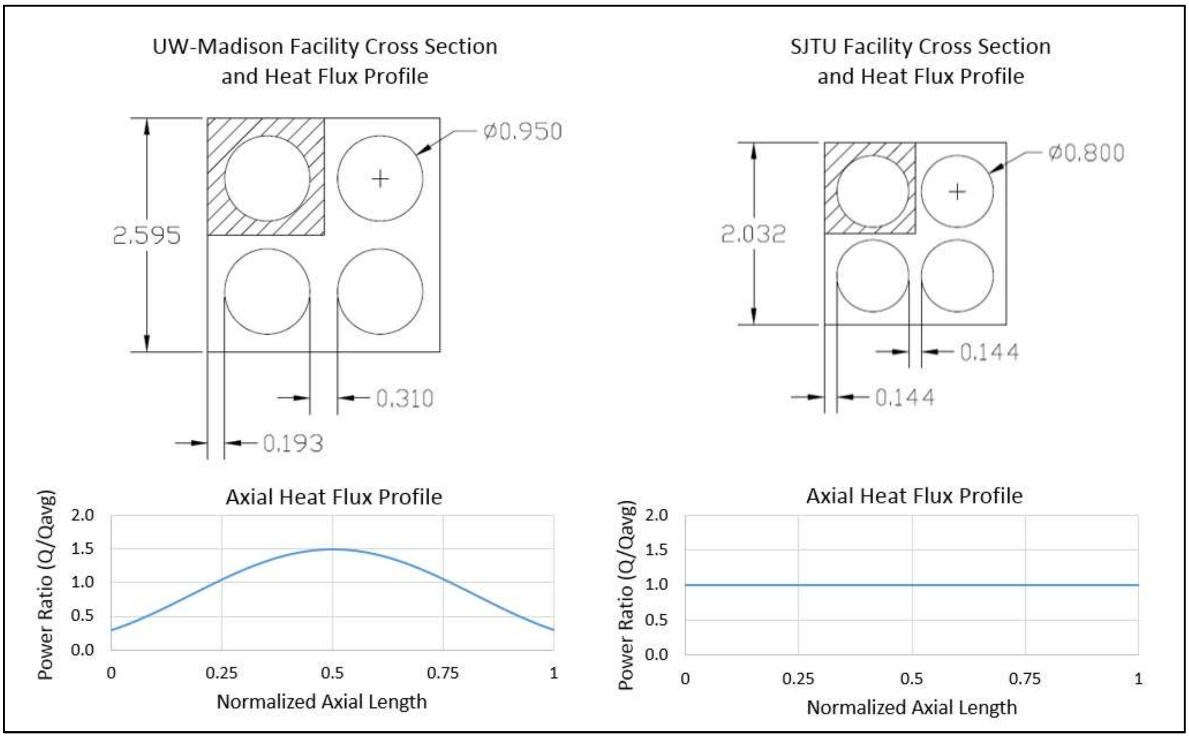

Figure 1: UW-Madison (left) and SJTU flow channel cross section (right) with dimensions in centimeters. The axial heat flux profile for the two facilities is shown below each cross section.

Two supercritical water experimental cases from the SJTU facility were chosen to model using CFD and the results were compared. Case A had an inlet temperature of 346.4°C and an inlet pressure of 25.09 MPa with a mass flux of 844.52 kg/m^2/s and a surface heat flux of 800.2 kW/m^2 per heater rod. Meanwhile, Case B had similar inlet temperature and pressure conditions of 340.1°C and 25 MPa respectively with considerably lower mass flux and heat flux conditions of 451.2 kg/m^2/s and 551.6 kW/m^2 per rod respectively. Different simulation models were used for both cases and their results were compared to the experimental data. This comparison led to a better understanding of the requirements of a simulation model for use in supercritical fluids. Furthermore, simulation of the two experimental cases allowed for comparison of differing heat and mass flux on the modelling technique.

The simulation techniques developed during this part of the benchmark will be used to predict the heat transfer characteristics of the UW-Madison test section at supercritical conditions. These models will need to be able to provide predictions regardless of heat flux profile and pitch to diameter ratios. By performing these benchmarks between facilities, the resulting simulation techniques will provide much useful information to further the understanding of supercritical heat transfer phenomena.

Presentation File Name: 11_TM52353_TS-III-3_W-Wang.pdf

DIRECT NUMERICAL SIMULATION OF FLUID FLOW AT SUPERCRITICAL PRESSURE IN A VERTICAL CHANNEL

W. Wang[1,2] and S. He[1]

[1]Department of Mechanical Engineering, The University of Sheffield, Sheffield, S1 3JD, UK

[2]Scientific Computing Department, STFC Daresbury Laboratory, Warrington, Cheshire, WA4 4AD, UK

Fluids at a supercritical pressure are accompanied by strong variations of thermoproperties, buoyancy-influences and abnormal thermal developments, which have complex effects on turbulence and heat transfer, making predictions and modeling of such flows a difficult task. A direct numerical simulation of water at supercritical pressure in a vertical channel is carried out to study effect of variation of thermo-properties and buoyancy on turbulence and heat transfer. The two walls of a vertical channel flow are set to be constant but different temperatures to isolate effects of variable properties and buoyancy from intricate thermal and flow developments. Under this condition, heat input from the heating wall and heat removal from the cooling wall are balanced to finally achieve a fully developed state with no heat advection but only thermal diffusion statistically. A new perspective on variable property and buoyancy effects is proposed, in which the phenomena are decomposed into bulk effect of variable property, density magnitude effect, density gradient effect, buoyancy effect interacted with density magnitude and buoyancy effect interacted with density gradient. The importance and impact mechanism of each element are compared and discussed. Turbulent and thermal characteristics near the cold and hot walls are discussed with regard to velocity, temperature, turbulence heat flux, turbulence statistics and turbulence productions. Turbulent flow structures are also investigated via the Lumley triangle analysis and iso-surface visualization of instantaneous flow structures.

Presentation File Name: 12_TM52353_TS-III-4_L-Wang.pdf

DEVELOPMENT OF THREE DIMENSIONAL CODE SYSTEMS FOR SCWR CORE

WANG Lianjie
Science and Technology on Reactor System Design Technology Laboratory, Nuclear Power Institute of China, Chengdu, 610041, China

In supercritical water cooled reactor (SCWR), huge change of the coolant density changes the core slowing-down cross-section field and has an important influence on the power distribution, and forms the complicated coupling characteristic between neutronics and thermal-hydraulics. The coupling between neutronics and thermal-hydraulics must be considered in core steady design, and also the spatial power distribution and its change must be considered for core transient analysis and safety evaluation.

The paper mainly focuses on the study of coupling three dimensional neutronics and thermal-hydraulics simulation for SCWR core steady state analysis and transient analysis. Coupled three dimensional core steady state analysis code and transient analysis code are developed and preliminarily validated.

The coupled three dimensional N/T-H code SNTA (**S**CWR coupled **N**eutronics/ **T**hermal-hydraulics **A**nalysis code) is developed for SCWR core steady state analysis. The cross section fitting module is improved for SCWR, and the in-core fuel management code NGFMN_S is developed based on three dimensional Nodal Green's Function Method. NGFMN_S and sub-channel code ATHAS is then modularly coupled, and the appropriate outer iteration coupling method and self-adaptive relaxation factor are proposed for enhancing convergence, stability and efficiency of coupled N/T-H calculation. By comparing the existing steady state code for SNTA verification, the numeric results show that SNTA is accurate and efficient for SCWR core steady state analysis.

Then the coupled three dimensional N/T-H code STTA (**S**CWR **T**hree dimensional **T**ransient **A**nalysis code) is developed for SCWR core transient analysis. Nodal Green's Function Method NGFMN_K is used for solving transient neutron diffusion equation. The SCWR sub-channel code ATHAS is integrated into NGFMN_K through the serial integration coupling approach. The dynamic link libraries method is proposed for coupling computation for SCWR multi-flow core transient analysis. The reliability and applicability of STTA are well proved by the PWR benchmark problem and SCWR rod ejection problems.

Presentation File Name: 13_TM52353_TS-III-5_Liu.pdf

THERMAL-HYDRAULICS CODE DEVELOPMENT IN SJTU FOR SCWR

X. J. Liu
School of Nuclear Science and Engineering, Shanghai Jiao Tong University

Due to its potential for high thermal efficiencies and considerable plant simplifications, supercritical pressure water cooled reactors (SCWR) have attracted strong interests of the international nuclear community and is recommended as the only water-cooled reactor among the six GEN-IV concepts. In the recent years, the application of supercritical fluid arouses lots of attention in the nuclear R&D filed due to its application to the SCWR. As one of the most important research institutes studying SCWR, Shanghai Jiao Tong University (SJTU) plays an important role in the supercritical fluid field. In the present paper, fundamental studies, both experimental and numerical, on heat transfer are carried out, to provide basic information for understanding the heat transfer mechanisms, to develop methods for code development for safety analysis of SCWR. The main achievements can be summarized as the followed fields:

(1) Perform CFD analysis for the supercritical fluid in subchannels, and develop new turbulence model to improve the predictive capability for heat transfer and mixing behavior;
(2) Improve some key models in subchannel code for supercritical fluid, e.g. mixing, heat transfer, pressure drop, and carry out the validation work by experimental data;
(3) Improve the current codes and perform the safety analysis for the current SCWR design by using the system code and subchannel code, to demonstrate the feasibility of the passive design system in SCWR.
(4) Couple the system code and subchannel code and perform the safety analysis for the SCWR-FQT facility.

The results achieved in this project can contribute to understanding the basic thermal-hydraulic phenomenon and to improve the accuracy of the current prediction method and SCWR design.

Keywords: supercritical fluid, thermal hydraulics, supercritical water cooled reactor (SCWR), model development

Presentation File Name: 14_TM52242_TS-I-1_Zheng.pdf

MATERIALS R&D FOR SCWR IN CANADA

Wenyue Zheng

Supercritical Water-cooled Reactors (SCWR) is one of the Gen IV reactor systems being developed through an international treaty-level collaboration led by Generation IV International Forum (GIF). Following the release of the core concepts from the Japanese and the EU members, Canada published its own pressure-tube based conceptual design in 2015. With an outlet temperature of 625 C and a core pressure of 25 MPa, the Canadian SCWR concept requires cladding materials that can sustain extremely harsh in-core physical and chemical conditions. Based on initial calculations using stainless steels and nickel alloys, the maximum cladding surface temperature is predicted to be as high as 825 C; the irradiation dose can reach as much as 10 dpa and the supercritical coolant can be very oxidizing due to the production of oxygen and hydrogen peroxide by radiolysis of light water. Under these conditions, the cladding can be readily degraded by corrosion, stress-corrosion, creep, or any of the radiation-related processes such as void-swelling and embrittlement. In the course of the Canadian program (2007-2015) on R&D of in-core materials, unique experimental and computational facilities were set up specifically to probe into the behaviours of candidate alloys under these extreme conditions. Some surprising results on corrosion and SCC have been achieved and the new insights will be valuable in guiding the design and development of new alloys.

Key highlights of this collaborative R&D effort are presented in this paper and the challenges for the future are also discussed.

Presentation File Name: 15_TM52242_TS-I-2_Penttila.pdf

SCWR MATERIALS RESEARCH AT VTT

S. Penttilä & A. Toivonen

VTT Technical Research Centre of Finland Ltd.

Nuclear Safety

Kivimiehentie 3, Espoo

P.O. Box 1000

FI-02044 Finland

The Supercritical Water Reactor (SCWR) has been selected as one of the candidate concepts for the new generation (GenIV) of nuclear reactors. Other than the design concept itself, the choice of construction materials is possibly the most challenging technical issue. As an evolutionary step from existing LWRs, it follows the development path of modern coal-fired power plants towards supercritical pressures and steam temperatures of up to 650°C. The objectives of the work were to assess the performance of potential candidate materials under the SCW environment in terms of SCC susceptibility, creep and oxidation resistance. Based on results, the main application for ferritic/martensitic (F/M)-steels will be the reactor pressure vessel (RPV) and ex-core components, like piping where temperature is low enough (≤ 400°C). The austenitic stainless steels were studied for the internals and fuel cladding. ODS (Oxide Dispersion Strengthened) steels may be an alternative to replace austenitic steels at high operating temperatures but some key challenges in manufacturing processes needs to be overcome. It is clear that it is not cost efficient to select the most alloyed material, e.g. Ni-based alloys, therefore, different techniques have been studied also to improve the oxidation resistance of traditional low alloyed austenitic steels, e.g. coatings and cold working. Ni-based alloys are also problematic to use as core components, since their high Ni content negatively affects core neutronics. In order to estimate oxidation rates, reliable autoclave data have to be obtained and models need to be developed and validated. For this purposes, a combination of ex-situ analytical studies of the oxide film forming processes with modelling approaches was proposed. An essential part of this work was also to develop testing devices and monitoring tools capable of working at SCW conditions. This presentation highlights the key findings in materials research performed in SCW during last 6 years at VTT. Most of the work has been performed in the following projects: EU HPLWR (High Performance Light Water Reactor), EU GETMAT (Generation IV and Transmutation Materials), EU SCWR-FQT (Fuel Qualification Test) and Academy of Finland projects NETNUC (New Type of Nuclear Reactors) and IDEA (Interactive Modelling of Fuel cladding Degradation Mechanisms).

Presentation File Name: 16_TM52242_TS-I-3_Novotny.pdf

SUMMARY ON SCWR MATERIALS RESEARCH ACTIVITIES PERFORMED IN EC-JRC

Radek Novotny

This work is a summary of experimental activities on SCWR materials research performed in EC-JRC in 2014-2016. The common objective is selection of fuel cladding and internals materials including characterization of general corrosion resistance of pre-selected austenitic stainless, characterization of sensors, characterization of SCC resistance of pre-selected materials and development of tools for future qualification testing. The achieved results are presented and discussed. At the end future plans are introduced.

Presentation File Name: 17_TM52242_TS-I-4_Starflinger.pdf

THE MUTUAL INFLUENCE OF MATERIALS AND THERMAL-HYDRAULICS ON THE DESIGN OF SCWR – REVIEW OF THE RESULTS OF THE PROJECT "HPLWR PHASE 2"

Joerg Starflinger,
University of Stuttgart, Institute of Nuclear Technology and Energy Systems (IKE),
Pfaffenwaldring 31, D-70563 Stuttgart, Germany

Thomas Schulenberg,
Karlsruhe Institute of Technology, Institute for Nuclear and Energy Technologies (IKET),
Hermann-von Helmholtz Platz 1, D-76344 Eggenstein-Leopoldshafen

Core design of supercritical water cooled Reactors (SCWR) is strongly influenced by the variation of thermodynamic properties of supercritical water near the critical point. Operating at a design pressure of 25MPa with a heat up from 280°C to 500°C, for instance, changes the density of supercritical water smoothly and continuously from a liquid-like state to a gas-like state. However, this smooth and continuous transition is non-linear. From the thermohydraulic design point of view, this non-linearity provides a special challenge in a hot-channel analysis. By definition, a hot-channel is a sub-channel somewhere inside the core, where all peaking factors (core-wise, assembly-wise), all uncertainties (experimental, numerical) and allowances (for plant operation) are condensed into one channel. The hot channel factor is multiplied to the nominal heat-up within the average channel to gain the enthalpy rise in the hot channel. The properties in this hot channel may not exceed the operational design requirements.

The hot channel method is a conservative method, because all uncertainties and peaking factors are applied to one channel, only. From the design point of view of a nuclear reactor not being built up to now, conservative methods should be applied at the first stage providing a suitable safety margin right from the beginning of the design.

Applying the hot channel factor analysis to the expected heat-up of a SCWR resulted in the three-pass core design as analysed in the HPLWR Phase 2 project. The three passes resulted from necessity to mix the supercritical fluid during the heat-up inside the core to avoid hot streaks, which could challenge the cladding material. For the HPLWR project, a maximum cladding surface temperature of 630°C was selected. Three pass simply means that the flow passes the reactor core three times: upward flow in the core – mixing in an upper plenum - downward flow in a different section of the core – mixing in a lower plenum – upward flow in a different section and delivery of supercritical water of 500°C to the turbine.

Several options can be applied to reduce the number of passes through the core. All of them have in common that either the hot-channel factor will have to be reduced, or the operation limits will have to be increased.

The first option is to select a material of the cladding which can withstand higher maximum cladding surface temperatures (increasing the operational limit). The higher the temperature, the more the material tends to increased corrosion, stress-corrosion cracking, or creep. High chromium content or nickel-base alloys are a possible path to go forward; however, low mechanical strength and increased parasitic neutron absorption must be taken into account.

Another option is to mix the supercritical fluid already in the fuel assembly. In the HPLWR Phase 2 project, wire wrap spacers (known from the breeder design) were foreseen instead of grid spacers which are used in BWR and PWR. Such wire wrap spacers mix quite well, but could possibly lead to hot spots on the cladding surface, which must be avoided through a suitable design.

Although the flow is well mixed within an assembly, the non-uniform neutron flux (and power) profile across an assembly still results in a non-uniform heat-up of the supercritical fluid. In the HPLWR design, three smaller assemblies were grouped into one assembly cluster with common head- and foot piece. The grouping was selected because control rod devices from existing PWRs (incl. control rod drives) should be used and the allowed number of flanges in the closure head of the reactor was limited at the increased pressure. However, for flattening of the neutron flux and power profile, smaller fuel assemblies (like in BWRs) would be more appropriate, which are also more flexible for shuffling.

The last option would be to reduce the mean core outlet temperature. In this case, the hot channel does not exceed 500°C whereas the nominal channel will be heated up to 390°C, only. Using this option, the supercritical water cannot be delivered directly to the turbine because of the liquid-like state. In this case, the SCWR design would be close to a PWR design with a closed primary system and with steam generators.

These options should be further investigated for the deployment of a Supercritical Water Cooled Reactor.

Presentation File Name: 18_TM52242_TS-II-1_Walters.pdf

STATE OF THE ART IN CLADDING AND FUEL CHANNEL MATERIAL SELECTION FOR CANADIAN SCWR CONCEPT

L. Walters[1] and M. Edwards

Canadian Nuclear Laboratories, Chalk River, Ontario, Canada K0J 1J0

The Canadian Super Critical Water-Cooled Reactor (SCWR) concept requires materials to operate at higher temperatures than current Generation III water cooled reactors. Materials performance after radiation damage is an important design consideration. Materials that are both corrosion resistant and radiation damage tolerant are required.

Although the extreme conditions and the broad range of SCWR in-core operating conditions present significant materials selection challenges, candidate alloys that can meet the performance requirements under most in-core conditions have been identified. However, for all candidate materials, insufficient data are available to unequivocally ensure acceptable performance. This presentation summarizes the knowledge gaps regarding the performance of candidate in-core materials and suggests experiments and data needed to verify their viability. Research programs are to include out-of-pile tests on un-irradiated and irradiated alloys, together with in-pile tests in a SCW loop to be constructed.

Presentation File Name: 19_TM52242_TS-II-2_Wang.pdf

CONCEPTUAL DESIGN FOR SCWR MATERIAL IRRADIATION TEST IN NPIC

Wang Hai

(Nuclear Power Institution of China, China)

In order to choose the appropriate material for internal structure materials and fuel cladding materials of SCWR, after out-pile choosing test, it needs research the in-pile irradiation performance, check the character of these material in high temperature environment. This article introduced the facility, the technical capacity and the engineer experience of NPIC in material and fuel in-pile irradiation test. Then introduce the in-pile test plan of materials and fuel about CSR1000, which developed by NPIC. 2~3kinds of candidate internal structure materials and fuel cladding materials are planned to conduct the in-pile test during 2016~2019 in HFETR. The materials irradiation date from these tests and results of post-irradiation examination would be a essential reference for CSR1000 design. An in-pile fuel assembly irradiation test loop with supercritical water in the research reactor of NPIC is also in the plan. The loop will simulate the real operating condition and qualify the fuel assembly of CSR1000, check the character of fuel assembly.

At last, the article describes the concept design of in-pile test and irradiate rig for CSR1000 internal structure materials and fuel cladding materials.

Presentation File Name: 20_TM52242_TS-III-1_Huang.pdf

EFFECT OF HIGH TEMPERATURE STEAM EXPOSURE ON OXIDATION BEHAVIOUR OF NICRAL AND FECRALY

Michael McTaggart and Xiao Huang

Department of Mechanical and Aerospace Engineering

Carleton University

In this study FeCrAlY and NiCrAl coating samples were tested in steam at 800°C for 300 and 600 hours. The FeCrAlY became discolored rapidly, while the NiCrAl still maintained some metallic sheen after 600 hours. The weight change results suggest that more oxide formation took place on FeCrAlY than on NiCrAl. In particular, grain boundary oxide (Al_2O_3) formed on FeCrAlY surface upon exposure to steam after 300 hours. Further exposure caused more intragranular Al_2O_3 to form, in addition to magnetite formations on the grain boundary regions. For the NiCrAl samples, NiO formed after steam exposure for 300 hours. Spinel and $(Cr,Al)_2O_3$ were also found after 300 hours, along with very limited amounts of Al_2O_3. After 600 hours in steam, Al_2O_3 became well-developed on NiCrAl, and the coverage of spinel and Cr_2O_3 on the surface was reduced.

Presentation File Name: 21_TM52242_TS-III-2_L-Zhang.pdf

SCWR CANDIDATE FUEL CLADDING MATERIALS AND THEIR CORROSION BEHAVIOURS

Lefu ZHANG, Wenhua GAO and Kai CHEN
Shanghai Jiao Tong University
A332 School of Mechanical Eng. No.800 Dongchuan Road, Shanghai 200240, P. R. China

On current available data, unsatisfied general corrosion and stress corrosion cracking (SCC) properties still remain the major problems among the candidate materials proposed for making fuel cladding of a supercritical water cooled reactor (SCWR), mainly due to the hight temperature and corrosive water environment. Ferritic/martensitic (F/M) steels, austenitic stainless steels, nickel base alloys, alumina forming steels, and their oxide dispersion strenthened (ODS) materials have been studied during the past years. Austenitic stainless steels, type 310S, 316L/316Ti and 800H, have been short listed and studied intensively. More recent results confirmed their susceptibility to SCC at SCWR operating temperatures. Scientists are looking for better candidates for fuel cladding and trying to innovate new materials with improved strength and lowered corrosion rate in high temperature and pressure water environment. ODS is an effective method in promoting the strength of base metal while reducing its susceptibility to SCC. F/M and austenitic stainless steel based ODS steels have been prepared and studied by several institutions, showing promising properties at both mechanical and chemical aspects. However, intensive study is still needed for characterization of fuel cladding material for our test SCWR. Future tests are defined for characterizing and verifying the reliability of the short listed materials, and general corrosion and SCC tests are still the major research works under plan.

Keywords: Supercritical water cooled reactor, Corrosion, fuel cladding, general corrosion, stress corrosion cracking, water chemistry.

Presentation File Name: 22_TM52242_TS-III-3_Fulger.pdf

OVERVIEW OF MATERIALS FOR SCWR TESTED IN RATEN ICN

E.M.Fulger, M.Mihalache, L. Velciu, D. Petrescu

RATEN ICN Pitesti

Supercritical water reactor (SCWR) is one of the Generation IV reactors characterized by operation above the thermodynamic critical point of water. A preliminary design of an SCWR revealed an operating pressure of 25 MPa and a coolant temperature of up to 620^0 C at the core outlet. Operating above the thermodynamic critical point of water the SCWR offers many advantages, such as simplified design, smaller volume, and higher thermal efficiency compared to the current light water reactor (LWR). However, supercritical water (SCW) can be a very aggressive oxidizing environment, especially at higher dissolved oxygen contents, and oxidation rates in such service condition are significantly enhanced. In a highly oxidizing and high temperature environment, corrosion of component alloys is expected to be more serious than that in a typical boiling water reactor environment. It would be a quite challenging to maintain structural integrity in the SCWR environment,especially in the core region. The most promising structural materials for the SCWR are austenitic stainless steels and nickel-base alloys. Literature review showed that the most probable fuel cladding material may have an austenitic structure and contain high Cr concentration up to 22% or higher. Therefore, a systematic study on the corrosion behavior of structural materials is needed to ensure their safe application to nuclear reactor systems.

This paper present a part of the research program performed at RATEN ICN on oxidation in supercritical conditions of commercially available austenitic alloys.

The work was focused on investigating the oxidation behaviour, surface morphology and surface microstructure of some candidate materials: austenitic stainless steels (304L, 310S, 316L, 321) and Ni based alloys (alloy 800HT and alloy 718) in water at two different temperatures (550^0 C ; 600^0 C) and 25MPa pressure.

After exposure in supercritical conditions, the studied alloys were characterized using various methods: gravimetry, optical microscopy, scanning electron microscopy (SEM), / energy dispersive spectroscopy (EDS), and X-ray diffraction (XRD).

A comparison of the alloys provides interesting information. Oxidation was observed to be the mainly form of high-temperature water corrosion experienced by the tested alloys. In terms of mass gain, the 310S steel had the best behaviour comparatively with the other samples. The weight gain in supercritical water decreased with Cr content, in order: alloy 316L > 321> 304L > 718 alloy > 800HT > 310S. Generally, the austenitic alloys developed dual-layered oxide scale in supercritical water, consisting of Fe_3O_4 for the outer layer and spinels (Ni, Fe) Cr_2O_4 for the inner layer. The both Ni base alloys, Inconel 718 and Incoloy 800HT developed a duplex oxide layer consisting of a nickel /iron rich outer layer and a chromium rich inner layer. In case of Inconel 718 complementary to oxidation, pitting corrosion has been observed but in time, oxidation (which is more predictable) was dominant.

Presentation File Name: 23_TM52242_TS-III-4_Maderuelo.pdf

EFFECT OF INTERGRANULAR CARBIDES IN THE OXIDATION AND STRESS CORROSION BEHAVIOR OF ALLOY 690 IN SUPERCRITICAL WATER

A. Sáez-Maderuelo, D. Gómez-Briceño, C. Maffiotte, F. Perosanz
CIEMAT, Avda. Complutense 40. 28040. Madrid (Spain).

The nickel base alloy 690, which was designed as a replacement for the nickel base alloy 600, is a material widely used in the nuclear industry due to its optimum behavior to stress corrosion cracking (SCC) under nuclear reactor operating conditions. Because of this superior resistance to some degenerative processes, the alloy 690 has been proposed as a candidate structural material for the Supercritical Water Reactor (SCWR), which is one of the designs of the next generation of nuclear power plants (Gen IV).

The presence of intergranular carbides in the alloy 690 plays an important role in the resistance of this alloy to SCC in caustic environments. In spite of this improvement, the mechanism behind it is not yet understood. In addition to this, Arioka *et al.* [1] observed higher resistance to SCC in specimens without intergranular carbides. Therefore the role of the intergranular carbides in the A690 is still an open question.

Considering these results, the aim of this work is to study the oxidation and stress corrosion behavior of alloy 690 with and without intergranular carbides in deareated supercritical water (SCW) at 400 ºC and 500 ºC at 25 MPa in order to gain some insight into the understanding of the effect of these carbides in the resistance of this material. In this presentation preliminary results of an on-going work will be presented.

[1] Arioka, K., et al. (2008). "Dependence of Stress Corrosion Cracking for Cold-Worked Stainless Steel on Temperature and Potential, and Role of Diffusion of Vacancies at Crack Tips." Corrosion 64(9): 691-708.

Presentation File Name: 24_TM522242_TS-III-5_Liu.pdf

THE EFFECT OF DISSOLVED OXYGEN ON STRESS CORROSION CRACKING OF 310SS IN SCW

Jinhua Liu[1], GONG Bin[1], E JIANG[1], YONGFU ZHAO[1]

[1]*Reactor Engineering Research Sub Institute, Nuclear Power Institute of China, P.O.Box622-206, Chengdu, Sichuan, 610041, PR China*

Austenitic stainless steels have been widely used as the major structural materials due to excellent combination of mechanical properties and corrosion resistance in high temperature. 310 SS has been regarded as a most promising material of fuel cladding for SCWR. It is well known that SCC of austenitic stainless steels is influenced by water chemistry, such as oxygen. In order to obtain a deep understanding of the SCC behavior of 310 SS in SCW, more research is urgently needed.

The effects of dissolved oxygen content on the tensile properties and stress corrosion cracking (SCC) susceptibility of austenitic stainless steel 310 SS were studied by performing Slow Strain Rate Tensile (SSRT) tests. The SSRT tests were carried out in supercritical water at temperature of 620℃, a pressure of 25MPa, and a strain rate of 7.5×10-7 s-1. The dissolved oxygen content was included 0, 500, 1000, 2000, and 8000 ppb. The SSRT tests were performed in a supercritical environment corrosion testing machine system, which can control the DO concentration, pH and conductivity.

After the tests, strain–stress curves were analyzed to identify the mechanical properties of 310 SS. The morphologies of the side surface and fracture surface on the specimens were conducted by scanning electron microscope (SEM), and the chemical composition and structure of the oxide formed were examined by energy dispersive spectroscopy (EDS), in order to investigate fracture mode and to evaluate SCC susceptibility.

The results show that the elongation decreased dramatically with the increasing of dissolved oxygen concentration. Cracks on 310 SS were widely distributed over the whole gauge section and a brittle fracture mode was observed on the fracture surface. The Cr content the oxide layer on the surface showed significant increase with the increasing of DO concentration. The corrosion products formed on 310 SS were Fe/Cr oxide layers, and they are classified into two layers, an outer Fe-rich oxide layer and an inner Cr-rich oxide layer.

Presentation File Name: 25_TM52242_TS-IV-1_Q-Zhang.pdf

CHEMICAL COMPOSITION MODIFICATION OF 310 TYPE STAINLESS STEEL

ZHANG Qiang, TANG Rui, XIONG Ru

Nuclear Power Institute of China

No. 328, Section 1 of Changshun Road, Shuangliu district, Chengdu 610213, China

Key words**:** 310 stainless steel, Chemical composition modification, microstructure stability, carbide

Austenitic stainless steels (ASSs) 310S is a promising fuel cladding material for super-cooled water reactor (SCWR), due to high strength, corrosion-resistance, oxidation-resistance, low sensitivity to neutron irradiation and good processing ability. However, a large amount of coarse $Cr_{23}C_6$ particles would be precipitated on the grain boundaries of 310S, which results in the increase of brittleness and the deterioration of corrosion resistance. Minor additions of strong carbide-forming elements, such as Nb, Ti, Ta, Zr, could suppress the formation of $Cr_{23}C_6$.

On the basis of 310S (25Cr-20Ni-0.08C wt.%), the cluster-plus-glue-atom model is introduced to design new alloy by multi-element co-alloying (Nb, Ti, Zr, Ta, W) of 310S. This cluster model dissociates the solid solution structure into a cluster part and a glue atom part: the cluster is the nearest-neighbor polyhedron and glue atoms are located in-between the clusters. It is found that the stable FCC solid solutions generally correspond to the cluster formula of [CN12 cluster](glue atoms)$_{1\sim6}$, where the cluster is a cub octahedron with a coordination number of 12. The basic Fe-Ni-Cr ternary composition of 310S($Fe_{55.0}Cr_{24.7}Ni_{22.3}$ wt.%) is determined as the cluster formula [Cr-($Fe_{10}Ni_2$)](Cr_4Ni_2), where Cr represents Cr-similar elements (Cr,Nb,Ta,Ti,Zr) and Ni represents Ni-similar ones (Ni,Mn,C).

A new alloy Fe-24.6Cr-22.2Ni-1.01Mo-0.09Nb-0.09Ti-0.17Ta-0.05C is designed from this formula by Nb, Ti and Ta co-alloying. The alloy ingots were prepared by vacuum arc melting processing. These ingots were then solid-solutioned at 1200 °C for 1h, stabilized at 950°C for 0.5h, and aged at 800 °C for 24h. The experimental results indicate that after stabilization treatment, a large amount of (Nb,Ta)C nanoparticles with a size of 50-70 nm are distributed on the grain boundaries of the matrix, besides minor TiC and $Cr_{23}C_6$. After aging treatment, the MC nanoparticles dispersed in the inner-grains uniformly, with no change of the particle size; a few $Cr_{23}C_6$ particles precipitate on grain boundaries, with a size of about 1 □m. It is suggested that the addition of Nb, Ti and Ta can form fine TiC, (Nb,Ta)C particles and the coarse $Cr_{23}C_6$ particles are suppressed.