Super Critical Water Reactor (SCWR)

Preamble: *For a sake of homogeneity among all the system reports within this Annual Report, this chapter has been intentionally synthesized in a reduced number of pages. The full extended version of the 2019 SCWR system report with the complete list of publications can be uploaded on the GIF website.*

Main characteristics of the system

The SuperCritical Water-cooled Reactor (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure vessel concepts proposed first by Japan and more recently by a Euratom partnership and China; and a pressure-tube concept proposed by Canada. Other than the specifics of the core design, these concepts have many similar features (e.g. outlet pressures and temperatures, thermal neutron spectra, steam cycle options, materials, etc.). Therefore, the R&D needs for each reactor type are common; this enables collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of the high thermodynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and PR&PP are also possible and are being pursued by considering several design options using thermal and fast spectra, including the use of advanced fuel cycles.

There are currently three Project Management Boards (PMBs) within the SCWR System: System Integration and Assessment (provisional), Materials and Chemistry, and Thermal-hydraulics and Safety. Canada, China and Euratom signed the extension of the Project Arrangements for Thermal-Hydraulics and Safety as well as the Materials and Chemistry in 2017.

R&D objectives

The following critical-path R&D projects have been identified in the SCWR System Research Plan:

- System integration and assessment: Definition of a reference design, based on the pressure tube and pressure vessel concepts, that meets the Generation IV requirements of sustainability, improved economics, safe and reliable performance, and demonstrable proliferation resistance. An important collaborative R&D project is to design and construct an in-reactor fuel test loop to qualify the reference fuel design. As a SCWR has never been operated before, such generic testing is considered to be mandatory before a prototype reactor can be licensed.
- Thermal-hydraulics and safety: Gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed for validating thermal-hydraulic codes. The design-basis accidents for a SCWR have some similarities with conventional water reactors, but the difference in thermal-hydraulic behavior and large changes in fluid properties around the critical point compared to water at lower temperatures and pressures need to be better understood.
- Materials and chemistry: qualification of key materials for use in in-core and out-core components of both pressure tube and pressure vessel designs. Selection of a reference water chemistry will be sought to minimize materials degradation and corrosion product transport and will be based on materials compatibility and an understanding of water radiolysis.

Main activities and outcomes

System integration and assessment

Four SCWR core concepts with thermal spectrum have been proposed. Canada, EU and Japan have completed their concept development. China is continuing the development of core and plant concepts for their pressure vessel type thermal spectrum SCWR. The China Pressure vessel-type SCWR (named CSR1000) has the following characteristics: thermal neutron spectrum, light water as moderator, two flow-pass of coolant in core, direct once-through cycle. The reference CSR1000 has the 9X9 pin by pin fuel assemblies with center 5X5 pin taken by the water moderator box. Recently, the fuel assembly and core structure design are simplified. In the new design, the UO_2 fuel rods are set around the tube to get moderated homogeneously and sufficiently. MOX fuel rods are settled in the outer zone to match spectrum. No Water rod or solid moderator are needed. **Figure SCWR 1** presents the reference and new FA design. Aiming at the reactivity control requirements of the SCWR core and its strong nuclear thermal coupling characteristics, a new type of control rod loading design was invented to overcome the shortcomings of the traditional "checkerboard" control rod loading design method. The new loading method reduces the number of control rod drive mechanism arrangements, reducing the difficulty in designing the SCWR pressure vessel top cover, and simplifying the control rod operation management procedure. Two project proposals have been approved by the China Ministry of Science and Technology in 2019 to promote the China SCWR design. The two projects start from 2020 and end in 2022. The international review of China SCWR design is supposed to be completed during this period.

Figure SCWR 1. **SCWR Thermal Spectrum Core Concepts**

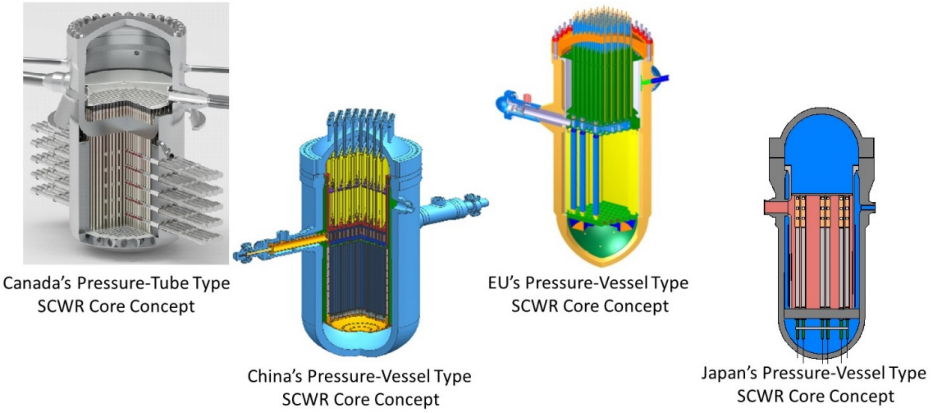
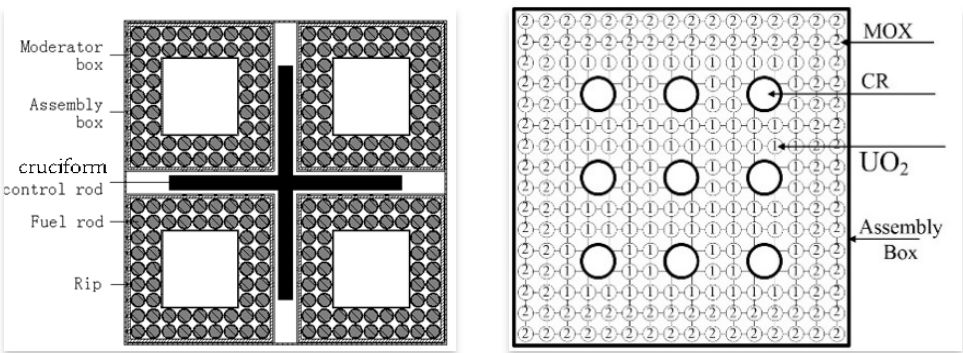


Figure SCWR 2. **China SCWR fuel assembly design improvement**

(a) Reference design (b) Improved design



Canada has developed a preliminary small pressure-tube type SCWR concept with similar core configuration. Currently the 43-Element bundle is the preferred candidate for the fuel bundle housing 170 fuel channels. The operating pressure 25 MPa with outlet temperatures about 450°C. Work on finalizing this concept is ongoing.

One of the main general activities covering almost all fields of the SCWR development was the preparation of the European-Canadian-Chinese Small Modular SC-Water Reactor Technology (ECC-SMART) proposal. The proposal joined the significant institutions working in the field of SCWR development from Europe, China, Canada and Ukraine to create very strong multi-international consortium. The proposal covers the major knowledge gaps in the thermal-hydraulics and safety and materials and chemistry issues for SCWR technology as well as the specific SMR topics related mostly to the scaling of the technology and legislation aspects.

The 9th International Symposium on SCWRs was held in Vancouver, Canada, March, 2019, hosted by Canadian Nuclear Society. The conference was supported by Natural Resources Canada, Atomic Energy of Canada Limited and Canadian Nuclear Laboratories. About 60 participants from Canada, China, EU and Japan attended this conference and 58 presentations covering concept development and technology areas were presented.

Thermal-hydraulics and safety

The TH&S PMB activities include flow and heat transfer experiments and correlations development, critical flow and flow instability investigation, numerical investigation and code development.

A project was established at the Canadian Nuclear Laboratories (CNL) to outline a framework of current prediction methods, the parameters associated with these methods, and the fluids they are applied to. The goal of this project is to complete a summary of literature reviews of heat transfer prediction methods in SuperCritical fluids (SC-fluids). From the literature review, the heat transfer to SC-fluids is dependent on at least six system parameters and four fluid flow parameters. The six system parameters include geometry, flow orientation, fluid flow direction, fluid type, heat flux direction, and power profile. The commonly used geometries and fluid types reflect the class of research and industrial applications where SC-heat transfer occurs. A more in-depth analysis of these parameters reveals that the majority of applications in which SC-heat transfer occurs is restricted to a narrow range of fluid parameters. These four fluid flow parameters are defined as: 1) Fluid pressure; 2) Fluid mass flux; 3) Surface heat flux; and 4) Fluid bulk enthalpy or bulk temperature. Six different SC-heat transfer prediction methods are currently used: 1) Correlations; 2) Semi-empirical models; 3) Look Up Tables; 4) Look Up Lists; 5) Neural Networks; and 6) Numerical/Computational Fluid Dynamics.

CNL conducted a study to investigate the applicability of a break discharge model that was specifically developed for supercritical conditions. To achieve this goal, the model was introduced in the Canadian thermal-hydraulics system code CATHENA. This model is hereafter referred to as the Modified Homogeneous Equilibrium Model (M-HEM). A comparison between the previously used homogeneous Equilibrium Model (HEM) and the Modified Homogeneous Equilibrium Model (M-HEM) model was performed. The assessment of the discharge models was performed by using experimental data in a simple geometry configuration (shows two representative results). The results of the assessment are used as a base to update the LOCA simulations used for the Canadian SCWR conceptualization.

The fuel channels of the Canadian SCWR undergo large density variation along the reactor core as condition of the coolant flow changes across the pseudo-critical point. To verify the stability of the design, CNL created a task aimed to verify, assess and develop a stability map for the Canadian SCWR design. This task was divided in two steps: i) assessment of the tools and ii) development of the Canadian SCWR stability map. Currently, CNL is focusing on pure thermal hydraulics instabilities and assessment of modelling tools. The tool selected was the system code CATHENA. Two datasets were selected to verify the applicability of the code: 1) the two parallel channel instability experiments carried out by NPIC (Nuclear Power Institute of China), and 2) the natural circulation numerical experiments conducted at the University of Manitoba. The simulation results showed that CATHENA is able to predict the flow oscillations, nonetheless the magnitude differs from the experimental data. However, given that the model

was simplified significantly, as recommended by the experimentalist, and the flow instability is highly dependent on geometry, this could have an impact on the simulation results. **Figure** SCWR4 shows two representative CATHENA prediction cases.

Figure SCWR 3. **CATHENA Predicted Coolant Mass Fluxes for 1-mm Orifice Diameter (Left) and 1.395-mm Orifice Diameter (Right) Supercritical Discharge Experiment at École Polytechnique de Montréal**

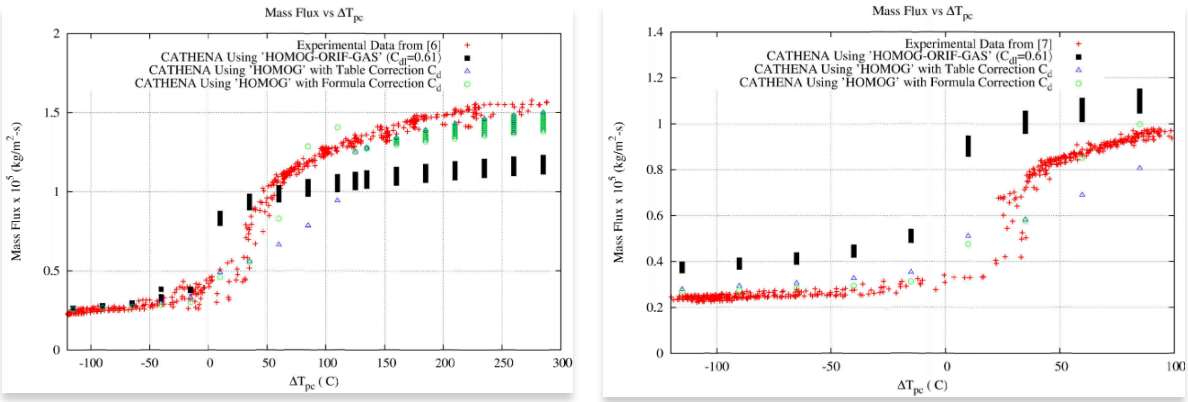
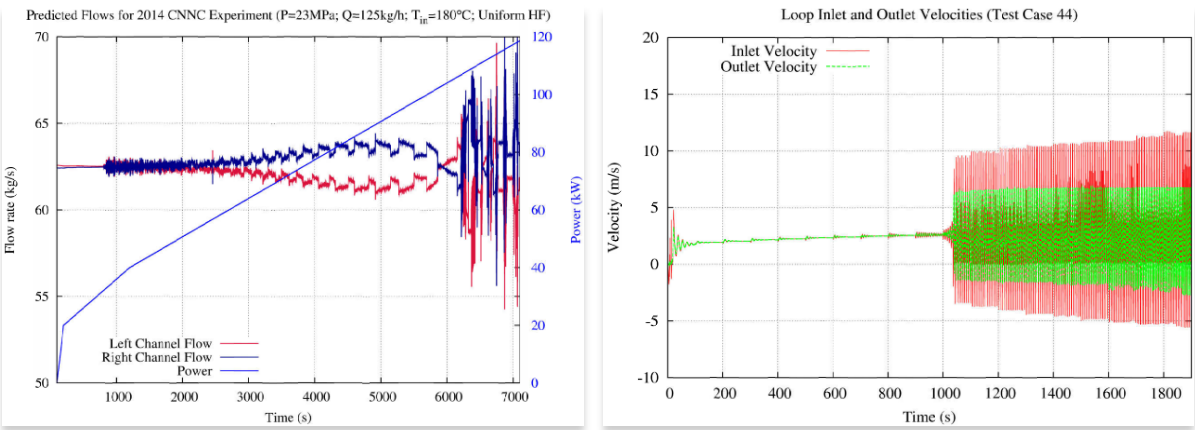


Figure SCWR4. **CATHENA Mass Flow Rate Predictions of Two Parallel Channel Instability (Left). CATHENA Predicted Velocities and Pressure Drops in a Natural Circulation Loop (Right)**



Heat transfer of in-tube SC-fluid cooling accompanying out-tube pool boiling was investigated in Xi'an Jiaotong University (XJTU). A smooth horizontal circular tube with an inside-diameter of 20 mm was submerged in a water pool at atmospheric pressure. Test parameters of in-tube were as follows: Pressure: 23-28 MPa, mass flux: 600-1 000 kg/m^{2}s, fluid temperature: 400-725 K, and the temperature difference between bulk and wall: 300–374 K. A thermal amplification system based on out-tube pool boiling was used to improve the measurement accuracy of local heat duty near pseudo-critical region. According to the experiment, the transition from nucleate boiling to film boiling in the pool occurred near the pseudo-critical fluid region. Sharp variation on thermo-physical properties led to the peak value of heat transfer coefficient in the pseudo-critical region. The pool boiling heat flux increased gradually to 1.19 MW/m^2 near the pseudo-critical point. Based on the experimental data, a modified Gnielinski equation was adopted to predict the heat transfer coefficient of in-tube SC-fluid cooling without-tube pool boiling.

Xi'an Jjiaotong University also performed experiments of heat transfer to supercritical Freon R134a flowing upward and downward in a circular tube with inner diameter of 10 mm with heat fluxes of 20-65 kW/m^2, mass fluxes of 400-1 000 kg/m^{2}s, bulk fluid temperature of 80-115°C at pressure condition of 4.2 MPa. The influences of heat flux, mass flux, flow direction, buoyancy force and flow acceleration on supercritical R134a heat transfer were discussed respectively. The influence of buoyancy force and flow acceleration on heat transfer were investigated and the non-dimensional parameters were obtained. New heat transfer correlations for upward and downward flow were proposed respectively.

Nuclear Power Institute of China (NPIC) performed Natural Circulation (NC) experiments and numerical analysis with water and carbon dioxide. For the supercritcal water NC instability, the preliminary analysis work has been done with the system analysis code. The code could predict the instability behavior of natural circulation. But some discrepancies exist which need further improvement. For the SC-CO_2 NC instability, based on the theoretical analysis of flow and heat transfer of SC-CO_2, a new explanation of the mechanism of flow oscillation in SC-CO_2 natural circulation has been put forward. The reliability of the new mechanism has been verified by experimental results.

China Institute of Atomic Energy (CIAE) performed the investigation of critical flow model for supercritical pressure condition. The model is derived to calculate discharge flow rate and critical pressure based on isentropic flow and thermal equilibrium assumptions. A correction coefficient of the influence of friction and local resistance is added. The model avoids the calculation of quality and is applicable to wide range which covering the subcooled water, two-phase mixture, steam critical flow under subcritical pressure and SC-pressure. The model calculated results agree well with the experimental critical flow data under SC-pressure.

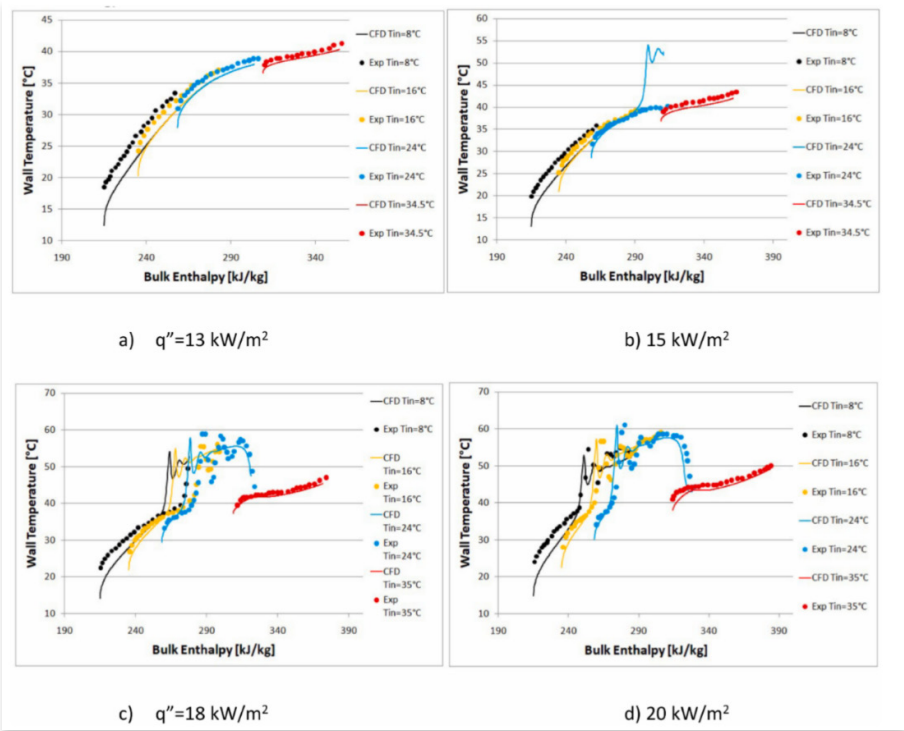
Numerical investigations have been performed by the researcher of BME NTI in 2019 in the field of SCWR TH&S. The thorium fueled SCWR concept was in the focus through a weakly coupled CFD – neutronics code-system. The thermal-hydraulics of the fuel assembly design of Th-SCWR has been simulated by the ANSYS CFX CFD code by a detailed 3D numerical model with and without wrapped wire spacer concept. The density field of SC-water (provided by the CFD results) was handed over to the MCNP Monte Carlo transport code as a boundary condition in each iteration step. The MCNP code calculated the field of heat source and this field was provided in return to the CFD code as a boundary condition. These calculations have proved that the wrapped wire spacer improves the heat transfer in most of the sub-channels within the fuel assembly and an axial and radial fuel enrichment distribution is essential for a viable fuel assembly design from the TH&S point of view. The linear heat source distribution has been optimized in the axial direction, but the maximum wall temperatures still seems to be higher than the melting temperature of currently available cladding materials. So, further optimization of the fuel enrichment is foreseen in the continuation of this research direction.

The CVR main activity on the thermo-hydraulic for Super Critical water Coolant (SCC) is based on the development and license activity that rotate in the insertion of the Super Critical Water Loop (SCWL) in the LVR-15 Reactor. For this reason, a consistent input deck of the facility was developed in ATHLET3.1A Patch1 code. From the several correlations adopted in ATHLET to simulate the SCC media, three were selected and qualified in Czech Republic: Watts-Chou, Mokry and Gupta. However, ATHLET3.1A Patch1 assessment was submitted to the thermo-hydraulic commission managed by the Regulatory State Office for Nuclear Safety (SONS) (Code and User Qualification) and it was qualified in March 2017. After the first revision of the flow regimes, all scenarios are reconsidered due to the lowering the operational pressure of the SCWL from 25 MPa to 24 MPa. The actual activity will focus in completing the flow regime scenarios according to the new specifications. Those selected scenarios analyses are used to verify the system performance in accordance with the safety criteria. A particular attention was given by providing operating regimes data in these conditions for structural analyses.

The University of Pisa developed RANS analyses of CO_2 data in 2017, making use of an Algebraic Heat Flux Model (AHFM) developed in the STAR-CCM+ code, on the basis of the Lien et al. model available in it. The RANS model is being assessed and improved on a variety of experimental data and the very systematic data by Kline offered the opportunity to understand capabilities and limitations of the improved AHFM as developed in this frame. The results, published in different steps showed a remarkable capability of the model to correctly simulate

heat transfer phenomena at relatively low flow rates. In particular, the phenomenon of deteriorated heat transfer termination at the transition to gas-like fluid was observed with reasonable accuracy, as shown in **Figure SCWR 5.**

Figure SCWR 5. **Results obtained for the cases with p=8,35 MPa, ID=4,6 mm and G=300 Kg/m2s**



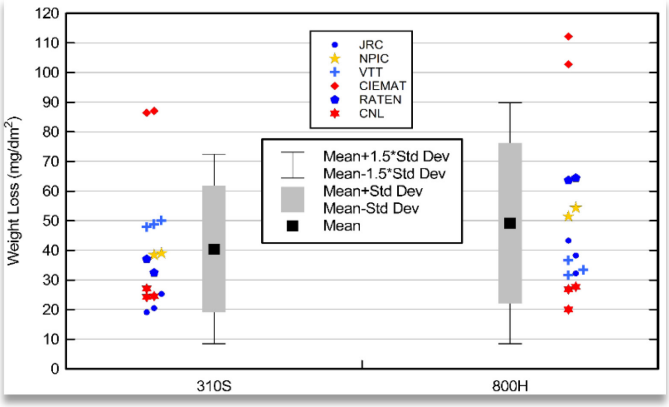
Materials and chemistry

The M&C PMB has been focusing on selection and qualification of candidate alloys for all key components in the SCWR. This includes general corrosion and stress corrosion cracking tests in autoclaves and loops as well as development work on test facilities, ionic irradiation tests of fuel cladding candidate alloy and development of novel alloys for fuel cladding. In addition, modelling of oxide film effects on fuel cladding heat transfer has been performed to better understand the interplay between general corrosion and thermal-hydraulics.

A major activity of the M&C has been the organization of a 2nd Round Robin corrosion test exercise between partners (Canada, China and Euratom) to compare the results of corrosion tests in different test facilities. Each laboratory used a standard test protocol with the coupon materials originating from the same batch and prepared by JRC-IEC. The tests were completed in 2017, and the coupons were sent to CNL for descaling. The results were reported in 2019. After 1 000 h exposure to 550°C supercritical water, coupons of Alloy 800H and Type 310S stainless steel were observed to have (descaled) weight losses of 54±26 mg/dm^2 and 41±22 mg/dm^2, respectively. Interestingly, the data were clustered by participant, with tight agreement between coupons of the same material exposed in the same facility at the same time, shown in Figure SCWR 6. It is not clear if the disagreement among participants is due to differences in flow velocity in the autoclave. It was proposed as a possible explanation.

In 2019, the Canadian materials and chemistry programme focused on expanding the set of high temperature general corrosion data, evaluating the effect of coatings on the corrosion of zirconium and titanium alloys in SC-water (500°C), and developing SC-water test facilities.

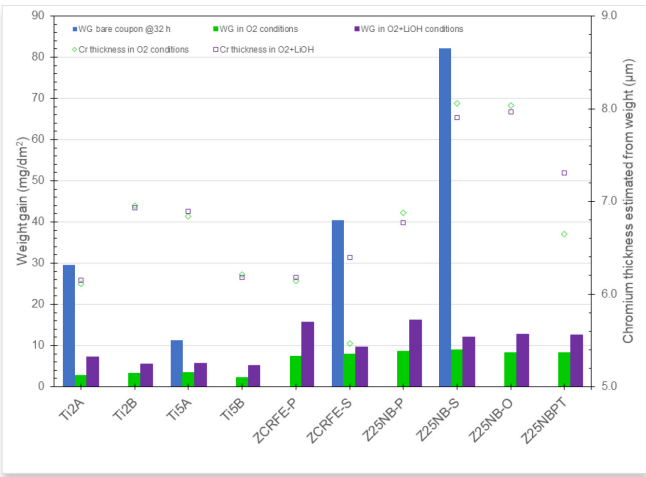
Figure SCWR 6. **Box plot of descaled weight loss data after 1 000 h exposure to deaerated water at 550°C and 25 MPa.**



Alloy 625 is a candidate fuel cladding for the benchmark Canadian SCWR concept with superior corrosion resistance. Experiments were conducted in 2019 that aimed to examine the effects of added hydrogen peroxide on the corrosion of Alloy 625 at temperatures between 650 and 700°C. These experiments also collected thermocouple data from a heated section of the loop that may be helpful in determining the effects of oxide growth on heat transfer. The collected data will be analyzed in 2020.

Corrosion testing of chromium-coated zirconium and titanium alloys was conducted in 2019 in support of the Canadian Small SCWR, a 300 MWe small modular concept. In an effort to improve neutron economy to lengthen the fuel cycle, coated zirconium alloys are being re-evaluated along with titanium alloys; the latter is an unlikely candidate given its poor neutron economy, but a Ti-50 enriched alloy would have good neutron economy and superior corrosion resistance. Coupons of Zr-1.2Cr-0.1Fe (R60804), Zr-2.5Nb (R60901 and R60904), pure titanium (R50400) and Ti-6Al-4V (R56400) were coated with a uniform layer of 10 µm chromium and exposed to 500°C oxygenated SC-water for 150 h, with and without LiOH as a pH-control agent. Weight gain measurements indicate an eight-fold improvement in corrosion resistance of coated coupons compared to the as-received alloys, shown in Figure SCWR 7. Alkaline treatment resulted in much higher weight gains than was found in pure oxygenated SC-water. Microscopic analysis of the coupons, as well as hydrogen uptake measurements, will be performed in 2020.

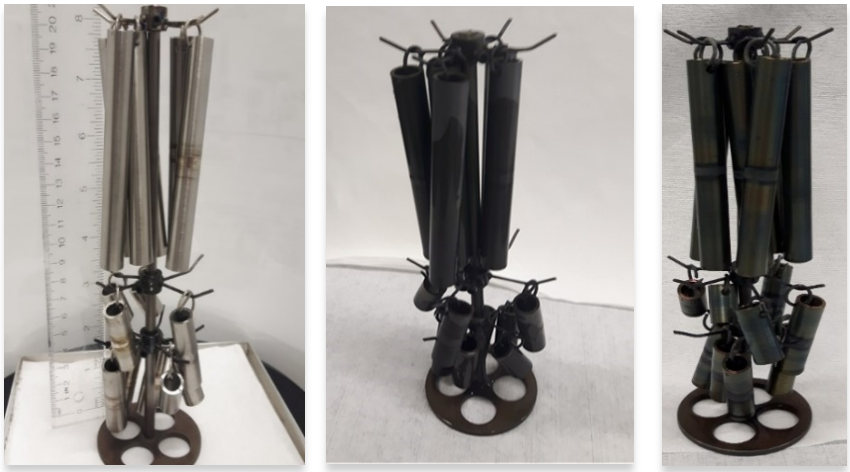
Figure SCWR 7. **Weight gain of chromium-coated zirconium and titanium alloys after exposure to 500°C SC-water for 150 h**



In addition, SC-water test facilities continued to be developed at CNL in 2019. A refreshed SC-water target system for a 2.5 MeV Van de Graaff electron accelerator was commissioned, and continues to be modified and improved. A SC-water hydrocyclone was designed with a MAWP of 29.25 MPa at 649°C (1 200°F) for future studies on activity transport and high temperature purification.

The corrosion and tensile behavior of both Alloy 625 and Alloy 800H welded specimens was examined by exposure to 575°C SC-water for 500 h at CNL. Tube specimens were butt welded autogenously by gas tungsten arc welding using a Swagelok® M200 orbital welding system. For Alloy 800H, post-exposure analysis indicated a 20% increase in the weight gain of welded specimens compared to unwelded specimens. Corrosion and aging of Alloy 800H reduced the ductility of both welded and unwelded specimens by 25% and increased the yield strength by 30%. For Alloy 625, which corrodes very little, welded specimens gained 60% more weight compared to unwelded specimens. Corrosion and aging of Alloy 625 reduced the ductility of both welded and unwelded specimens by 40% and increased the yield strength by 25%. Welded specimens that had been exposed to SC-water were observed to have 10% higher UTS and up to 15% lower ductility when compared unwelded specimens.

Figure SCWR 8. **Before Corrosion Test (left), 800H After 250 hours of Exposure (center) and Alloy 625 After 500 hours of exposure (Right)**

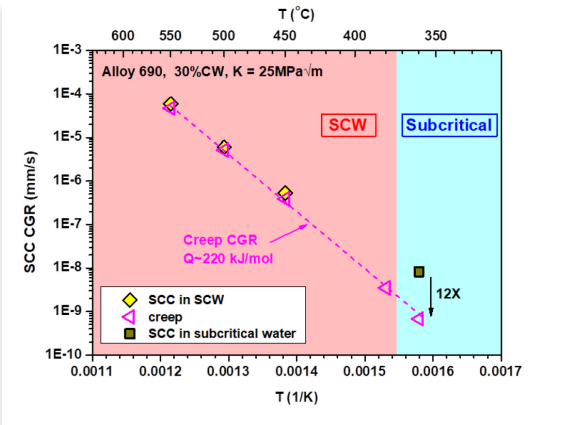


The SCC tests were conducted on 30% cold worked Alloy 690 in high-temperature pressurized water at temperatures between 360°C and 550°C in Shanghai Jiaotong University (SJTU). Creep induced cracking was measured at each testing temperature in inert argon environment to study the effect of creep on the overall crack growth behavior. The crack growth rates at each condition are summarized in **Figure SCWR 9.** Experimental results showed that creep contributed to more than 80% of the overall crack growth rate at temperatures above 450°C in SC-water, while only 8% at 360°C in subcritical water. It was clearly proved that the dominant mode of cracking from subcritical to SC-temperatures are different for cold worked Alloy 690. Corrosion induced cracking controls the crack growth rate in subcritical water environment, while creep is the major factor that dominates the cracking in SC-water.

The effect of InterGranular (IG) carbides on the cracking behavior of cold worked Alloy 690 was also studied in both subcritical and SC-water environments. The crack growth rate was lower when IG carbides were removed by prior solution annealed (SA) treatment, indicating a detrimental effect of IG carbides. The presence of IG carbides enhances the local strain accumulation at the grain boundary due to the lattice mismatch, thus promotes the crack tip strain rate and increases the crack growth rate. The SCC CGRs of Alloy 690 were compared with those of 310S SS in SC-water, and results show that the SCC CGR of the 310S SS specimen in 550°C SCW was 1.4×10^{-7} mm/s, ~17 times higher than the Alloy 690 specimen (8.3×10^{-9} mm/s) at

the same testing condition. The degree of sensitization prior to and after the test was confirmed by both double loop electrochemical potentio-kinetic reactivation (DL-EPR) method and TEM analysis at the grain boundaries. It was found that the degree of sensitization increased dramatically for 310S than Alloy 690 after the SCC test, indicating severe in situ sensitization occurred in 310S during the SCW exposure.

Figure SCWR 9. **Comparison of the SCC and creep CGRs of Alloy 690 in subcritical and supercritical water at the temperatures ranging from 360°C to 550°C**



Two candidate alloys modified from 310S austenitic stainless steels were subjected to ionic irradiation to see their radiation damage effects in Nuclear Power Institute of China (NPIC). The major difference in these two alloys lies in their optimized minor alloying elements, addition of Mo, Nb, W and Ta in alloy SC1, and in SC2, Mo and Zr were added. Proton radiation tests were performed on an ion accelerator at Wuhan University, with implanting energy of 50 keV, and temperature at 290°C up to doses of 0.1 and 0.3 dpa by proton, and at 550°C up to 5, 15 and 30 dpa by Ar ion. Figure SCWR 10 shows the TEM micrographs of specimens irradiated at 550°C and Figure SCWR 11 shows the defect caused by irradiation. The irradiation tests showed that minor alloy elements added to the alloys played different roles after irradiation. At 290°C, Zr modified alloy SC2 showed lower density of void and dislocation loop defects than SC1, which contains Nb, W and Ta. However, at 550°C Zr caused void swelling in SC2 while Nb and Ta in SC1 reduced the density of voids.

Figure SCWR 10. **The TEM micrographs of ally SC1 and SC2 irradiated at 550°C**

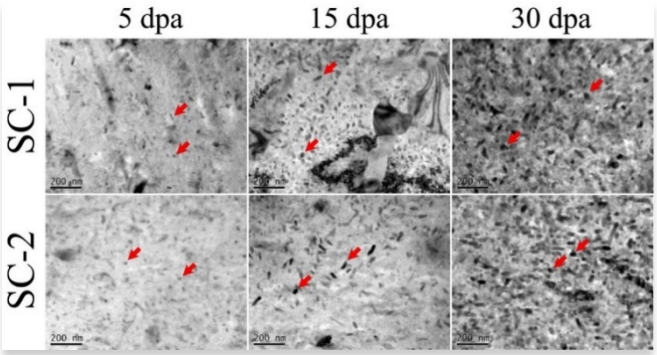
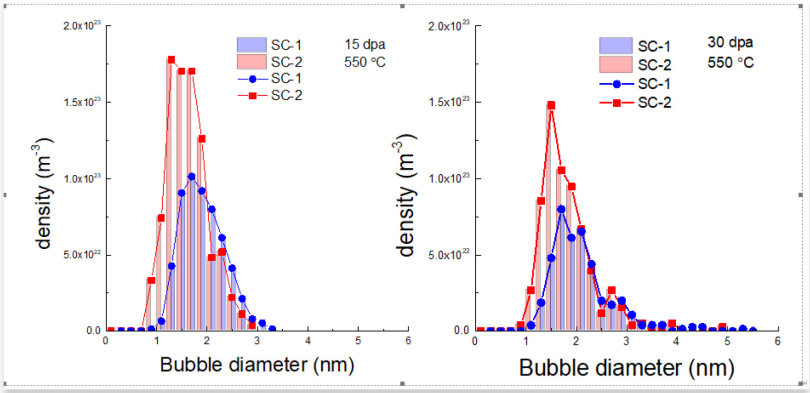
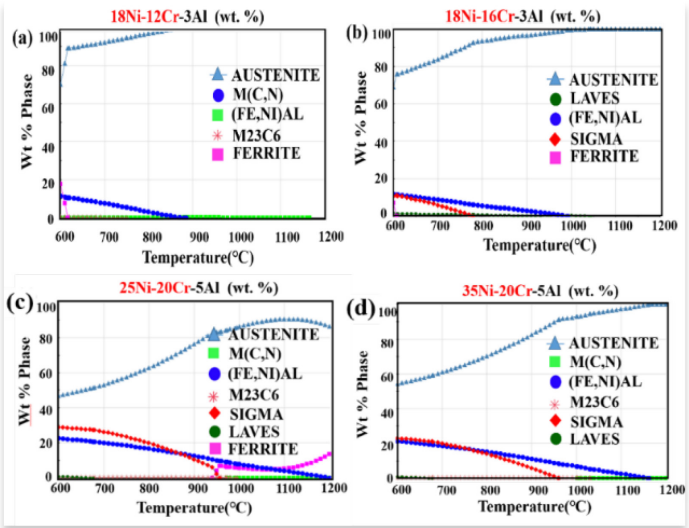


Figure SCWR 11. **The density of defects obtained by TEM of ally SC1 and SC2 irradiated at 550°C**



The major activity at the University of Science and Technology Beijing (USTB) has been focusing on developing of novel candidate alloys for cladding tube application. Alumina Forming Austenitic (AFA) alloy is proposed as a new grade SCWR candidate alloy in view of the existing reported research works, as well as results from the Round Robin corrosion test exercise between PMB partners (Canada, China and Euratom). One of the key challenges for the composition design of AFA alloys is to balance the corrosion resistance and maintain a single austenitic matrix phase for good creep strength. Figure SCWR 12 shows the matrix phases and fractional volume of precipitates in the materials with different Al% at temperatures between 600 and 1 200°C, which was calculated by using the computational thermodynamic calculation program, JMatPro. It is obvious that high Al, Cr content, but insufficient Ni content (25Ni20Cr5Al) will result in the formation of duplex $\gamma+\alpha$ matrix, as Al is a strong ferrite stabilization element. Therefore, the content of Ni should be carefully designed based on the content of Al, Cr and other minor ferritic forming elements to obtain a necessary single γ phase structure. ODS austenitic alloy is the other promising new grade material for the in-core structure application in SCWR. The microstructure stability of a 310 type ODS austenitic alloy after aging at 500°C for different length of time is investigated.

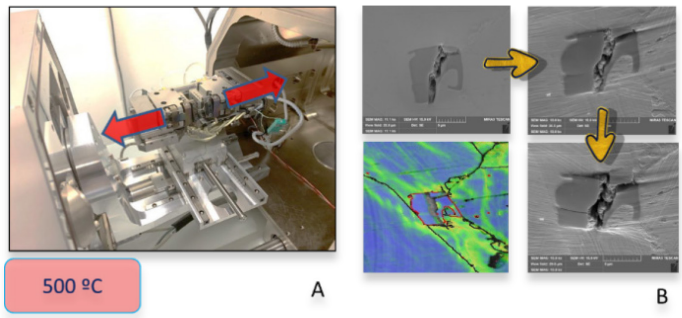
Figure SCWR 12. **Phases in AFA alloys with different Ni, Cr and Al contents based on Jmat- Pro calculations**



Along 2019 Ciemat has performed several in situ tensile tests with a nickel base alloy 690 TT previously tested in deaerated Supercritical Water (SCW) at 500°C. These tests were carried out in collaboration with CVR laboratories. The aim of this work was double, on the one hand, considering this was the first time the staff of CVR in Pilsen work with this material and this specimens geometry, the team tried to write a procedure to study the Alloy 690 TT by means in tensile in situ tests within a Scanning Electron Microscope (SEM) at high temperature (see **Figure SCWR 13**). On the other hand, the second objective was to follow the evolution of precipitates, defects, etc., in the microstructure by means the in situ test. As a result of this first attempt, both laboratories were able to follow the evolution of C,N(Ti) with temperature and strain. This test showed how the plastic deformation gathered around the carbides during the tensile test. This behavior, that has been reported by other groups, could have some meaning in the cracking processes of this material especially if the CNTi are near the surface in contact with SCW at high temperature. After these tests new tensile specimens were designed in order to avoid the problems detected during the first tests. These new specimens will be used to continue this work and to gain more in-depth knowledge into the mechanical behavior of some defects present in this alloy like the vacancies along the grain boundaries that appear as a result of cold work and high temperature. The main results from this work were presented in the EUROCORR congress that was held in Seville (Spain) and in the EPRI meeting EPRI Alloy 690/52/152 Primary Water Stress Corrosion Cracking Research Collaboration Meeting that was held in Tampa (United States).

In addition to this, Ciemat co-ordinates the European funded project MEACTOS in which there is a small task related to SCW. In this task the SCW will be used as an accelerating environment to produce stress corrosion cracks in an austenitic stainless steel 316 L type L.

Figure SCWR 13. **A) Photograph of the device used to perform the tensile in situ tests inside the chamber of a SEM: B) Evolution with temperature and stress of a C,N(Ti) found in a A690 TT specimen previously tested in SCW**



The activities at VTT in 2019 have mainly come through its participation in the EU MEACTOS project where SCW is used as accelerating method of the stress corrosion cracking processes in an austenitic stainless steel. Moreover, VTT participated in the writing of the ECC-SMART proposal for the Horizon 2020 call. In addition to this, VTT has prepared together with Aalto University a project proposal called TAMAT (Towards Advanced Materials for Energy Technologies: Multimetallic Layered Composites and Innovative Cladding Solutions for Nuclear and Beyond) to the Academy of Finland where one work package is dealing with experimental testing and oxide film modelling in supercritical water conditions. The Academy of Finland decision is expected by June 2020.

The M&C in CVR focused on microstructural evaluation of candidate materials for SCWR internals and fuel claddings. First three materials – 800H, T505 (uquivalent of T91) and 08Cr18Ni10Ti (equivalent of AISI 321) were exposed in the supercritical water loop(SCWL) in 2018. In the end of the 2019, next exposure in SCWL started with another three candidate materials (Nimonic 901, Nitronic 60 and In 718). First corrosion exposure up to 550 h and second exposure up to 1 000 h duration were carried out at 400°C/ 25 MPa with deoxygenated water, pH 6, conductivity under 2 µS/cm, Fe < 100 µg/l.

Materials from first exposure were analyzed by SEM technique in combination with EDX for chemical composition and EBSD for crystallography. Final Raman and XRD analysis confirmed compounds of magnetite oxides (Fe_3O_4) on all surfaces. No significant oxide layer occurred on 800H and 08Cr18Ni10Ti, only random not compact oxide particles. Double spinel (3-7 µm) layer occurred on T505: inner passivation layer of chromite $FeCr_2O_4$/trevorit $NiFe_2O_4$ and outer layer of magnetite Fe_3O_4. Other investigations in SCW were development works on two autoclaves with parameters: volume 137 ml, 600°C/25 MPa and volume 850 ml, 700°C/30 MPa. These autoclaves are supposed to work from 2020. One more autoclave is supposed to be developed in hot cells next year, to expose irradiated materials to SC-Water.

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