Supercritical water reactor

Main characteristics of the system

The supercritical water-cooled reactor (SCWR) is a high-temperature, high-pressure, water-cooled reactor that operates above the thermo-dynamic critical point (374°C, 22.1 megapascals [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 with China, and a pressure-tube concept proposed by Canada. Apart from the specifics of the core design, these concepts have many similar features. The R&D needs for each reactor type are therefore common, enabling collaborative research to be pursued.

The main advantage of the SCWR is improved economics because of the high thermo-dynamic efficiency and the potential for plant simplification. Improvements in the areas of safety, sustainability, and proliferation resistance and physical protection 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: for system integration and assessment (provisional), materials and chemistry (MC), and thermal-hydraulics and safety (THS). The extension of the project arrangements to thermal-hydraulics and safety, as well as to MC, with project plans covering 2021-2025, are in progress and have been discussed during the steering committee and Project Management Board meetings held in early December 2020.

R&D objectives

The following critical-path R&D projects have been identified in the SCWR system research plan (SRP):

- System integration and assessment: definition of a reference design, based on the pressure tube and pressure-vessel concepts, which meets Gen-IV requirements (sustainability, economics, safe and reliable performance, proliferation resistance).
- Thermal-hydraulics and safety: gaps exist in the heat transfer and critical flow databases for the SCWR. Data at prototypical SCWR conditions are needed to validate 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.
- MC: 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 on an understanding of water radiolysis.

Main activities and outcomes

System integration and assessment

Because of the COVID-19 pandemic, and priority work on the THS and MC PMBs, no GIF SCWR activities were undertaken in this field in 2020.

The Joint European Canadian Chinese Development of Small Modular Reactor Technology (ECC-SMART) project was launched in September 2020. ECC-SMART is a collaborative project covering most GIF SCWR research fields. ECC-SMART is oriented towards assessing the feasibility and identification of safety features of an intrinsically and passively safe SMR cooled by supercritical water (SCW-SMR). The project takes into account specific knowledge gaps related to the future licensing process and the implementation of this technology. The main objectives of the project are to define the design requirements for the future SCW-SMR technology, to develop a pre-licensing study and guidelines for demonstration of safety in the further development stages of the SCW-SMR concept, including the methodologies and tools to be used, and to identify key obstacles for future SMR licensing and a strategy for this process.

To reach these objectives, specific technical knowledge gaps were defined and will be assessed to better achieve future licensing and implementation of the SCW-SMR technology, particularly in terms of the behavior and irradiation of materials in the SCW environment, and validation of the codes and design of the reactor core, evaluated through simulations and experimentally validated.

The ECC-SMART project consortium consists of the EU, and Canadian and Chinese partners, who are making use of trans-continental synergy and knowledge developed separately by each partner, as well as under the GIF umbrella. The project consortium and scope were created according to joint research activities at the IAEA and at GIF, and as much data as possible will be taken from projects already performed. ECC-SMART brings together the best scientific teams working in the field of SCWRs, using the best facilities and methods worldwide, to fulfil the common vision of building an SCW-SMR in the near future.

For China, two projects supported by the Ministry of Science and Technology of China (MOST) were started in 2020. One is the GIF SCWR THS and the other is the GIF SCWR MC. The main goals of the two projects are to improve the China CSR1000 design and finish the international review before the end of 2022 so as to compile an expanded database based on previous research results. A kick-off meeting was held in 2020 in the Nuclear

power Institute of China (NPIC), Chengdu, China. Five Chinese institutes participated in the two projects, including the NPIC, Shanghai Jiaotong University (SJTU), Xi'an Jiaotong University (XJTU), the China Institute of Atomic Energy (CIAE) and the University of Science and Technology Beijing (USTB). Two virtual meetings were held online in March 2020 and September 2020 as a result of the COVID-19 situation.

Thermal-hydraulics and safety

Euratom activities

The ECC-SMART project was launched in September 2020, and it comprises several different work packages (WP). WP 3 in particular focuses on thermo-hydraulics and safety analyses, and the current task is to define a common design that is to be analyzed.

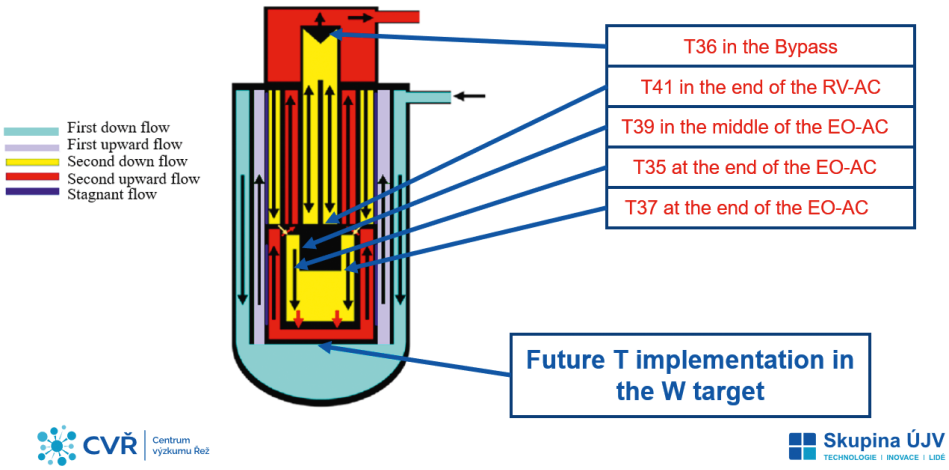
In Hungary, two institutions are working in close collaboration on THS research for SCWRs: the Centre for Energy Research (EK) and the Budapest University of Technology and Economics (BME). The EK continued its experimental activities (not only on heat transfer) in collaboration with the Department of Energy Engineering (DEE), the Department of Chemical and Environmental Process Engineering (DCEPE) and the Institute of Nuclear Techniques (NTI) of the BME. Two experimental works have been proposed in 2020 and may be elaborated further in the near future. The BME DEE has been working on theoretical research of water chemistry and thermal-hydraulic issues related to SCWs during the 2020 year. The BME NTI continued its numerical and theoretical research on the thermal-hydraulics of SCWs.

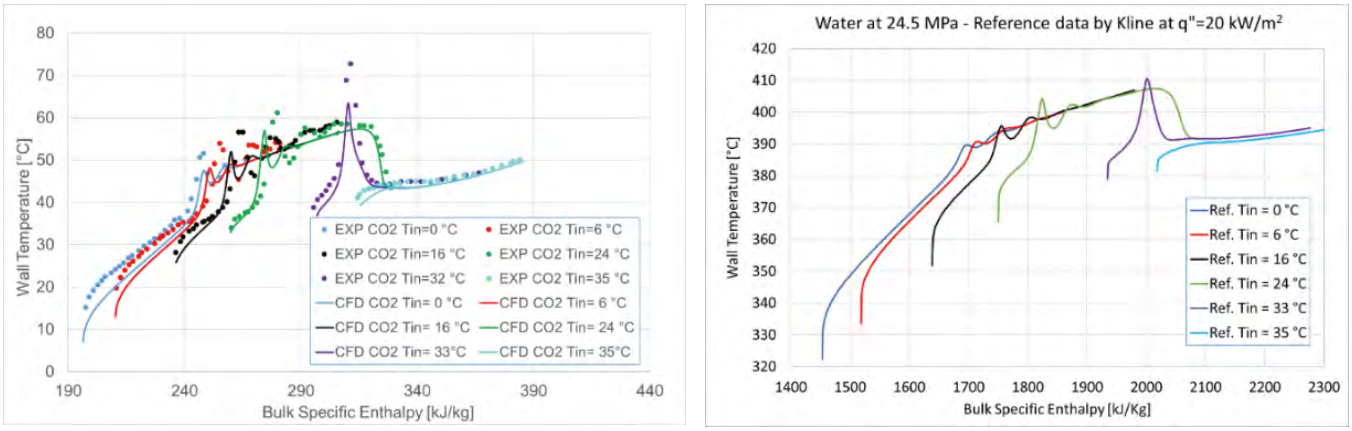
The main activity of the Research Centre Řež focused on the second licensing phase in order to insert the supercritical water loop (SCWL) in the LVR-15 reactor (Musa et al., 2020). This advanced facility is monitored by the Czech Republic State Office for Nuclear Safety (SONS). New analyses were performed with the goal of providing boundary conditions for the assessment of stress and strain calculations. The analyses were performed in ATHLET 3.1 code. After the initial revision of the flow regimes, all scenarios were reconsidered as a result of the lowering of the SCWL operational pressure from 25 MPa to 24 MPa. In addition, a new criterion for a new SCRAM signal will be implemented. The design will be improved by inserting a thermocouple in one of the two tungsten targets (T SCRAM around 550°C). This thermocouple could provide additional information on the heat transfer during transient phase.

Additional activities focused on simulating the SCWR thermo-hydraulic behavior. ATHLET code was benchmarked with the experimental data from the out-of-pile configuration. Among the postulated scenarios, an abnormal sequence (labelled A2 – loss of power in the loop) was analyzed. This scenario is similar to the postulated in-pile A2. The analyzed correlation in this phase were performed by Gupta et al. and Mokry et al.

In the past years, the University of Pisa in Italy has been addressing by a RANS model CO_2 data, water data and other data produced by several researchers, making use of an algebraic heat flux model (AHFM) developed in the STAR-CCM+ code, on the basis of the Lien et al. (1996) model. The RANS model was assessed and improved on a variety of experimental data, obtaining good results in comparison with experimental data. Based on these results and on data provided through direct numerical simulation (DNS) studies by a group at the University of Sheffield, in 2020 the subject of a fluid-to-fluid similarity theory for heat transfer at supercritical pressure (already proposed in past years) was further developed. Very good results were obtained as a result of these new steps, making it possible to confirm a sound rationale for assessing the scalability of results obtained with different fluids. The results of this work, performed at a distance during the current pandemic, have been published in three papers produced in 2020 and 2021 (Pucciarelli et al., 2020; Pucciarelli and Ambrosini, 2020; Kassem et al., 2021). Figure SCWR-2 provides comparisons between the original experimental data by Kline

Figure SCWR-1. Active channel thermocouples map





Note: as predicted by the similarity theory (data by Kline: CO_2, 8.35 MPa, 4.6 mm ID, q"=20 kW/m2, G=300 kg/m2s, upward flow, different inlet temperatures).

Figure SCWR-2. Comparison between experimental and data (right) and corresponding similar water trends (left)

obtained with CO_2, with the corresponding trends predicted in fluid similarity in the case in which water would be used instead, providing a clear idea of corresponding trends that can guide experimentalists in planning their experiments with simulant fluids.

SCWR research at the University of Sheffield (USFD) focuses on high-fidelity numerical simulations using DNS to produce high-quality reliable data in order to complement physical experiments. These tend to be for lower Reynolds numbers, but are able to produce detailed information and data to help understand the physics and the development of practical engineering models. The USFD has developed a versatile DNS code, CHAPSim, which has now been selected by the UK Collaborative Computational Project for Nuclear Thermal Hydraulics – supporting next generation civil nuclear reactors (CCP NTH) (sponsored by EPSRC EP/T026685/1 2020-2025) to be developed as a UK NTH community code. Over 2020, USFD work has included: 1) implementing conjugate heat transfer in CHPASim and carrying out some preliminary simulations; 2) carrying out simulations of flows in a horizontal orientation; and 3) investigating the development of a unified approach to explain the mechanisms of flow laminarization and heat transfer deterioration in a heated vertical flow of supercritical fluid. The USFD has also started working on simulations of SuperCritical fluid flows over rough/corroded surfaces in the context of the ECC-SMART project.

Activities in China

Two supercritical water thermal-hydraulics benchmarks were released in November 2020 in China. One is on the 2X2 bundle SC-water tests performed by the NPIC and the XJTU several years ago. The other is on parallel channel instability experiments of supercritical water. The structure layouts are shown in Figure SCWR-3. The 2X2 rod bundle tests are used to validate the computational fluid dynamics (CFD) tools (e.g. CFX, Fluent, Star CCM+, Open-FOAM) and subchannel tools (e.g. ATHAS, SC-COBRA). The parallel channel instability experiments were used to validate the system analysis tools, such as SC-TRAN, RELAP5 and APROS. The first-round comparisons are planned for 2021.

Experiments on heat transfer of supercritical water in a single subchannel with grid spacer were successfully carried out. More reference characteristics about the influence mechanism of the grid spacer can be observed through this experiment in Figure SCWR-4. The section structure of the test part is clearly shown in in Figure SCWR-3. The test section is a triangular-shaped sub-channel with a standard grid spacer of an 8 mm core rod diameter, 1.4 pitch ratio, 990 mm total length and 2.5 mm thickness. The grid spacer was located 550 mm from the inlet. The influence of pressure, flow rate, and heat flux on the standard grid spacer within the framework of sub-channel flow heat

Figure SCWR-3. The structure layout of 2X2 rod bundles (left: from NPIC; right: from XJTU)

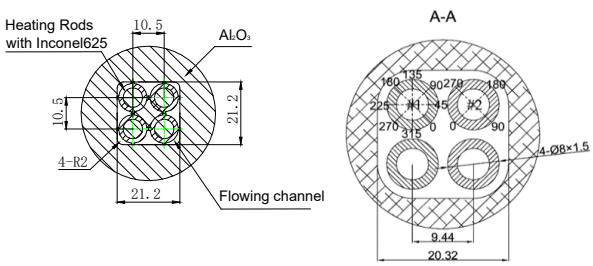
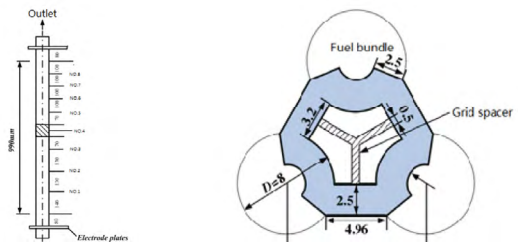


Figure SCWR-4. The structure layout of the sub-channel test section



transfer characteristics of supercritical water were studied. Figure SCWR-3 shows the distribution curve of the heat transfer coefficient, changed with the enthalpy in several working conditions. With the comparison of the five curves, the grid spacer greatly influenced the heat transfer characteristic of the entire flow, especially downstream. For the downstream flow direction, the fluid disturbance was reduced because of the decreasing blockage area. The circulation area abruptly widened in comparison to the grid spacer.

A new thermal amplification system was constructed to improve the accuracy of the calculated heat transfer coefficient of a supercritical fluid near its pseudo-critical point with the influence of the parameters of supercritical water cooling via downward flow inside a tube accompanied by pool boiling outside of the tube (McLellan et al., 2021). The results of this study will be helpful in understanding the effects of pressure, fluid temperature, mass flux and heat flux on the characteristics of supercritical downward cooling heat transfer. The test section is shown in Figure SCWR-6. A type 316 stainless steel circular tube was used as the test section. Its inner diameter and wall thickness were 20 mm and 2.5 mm, respectively. Five-sixths of the test section was directly immersed in the water of test pool. Figure SCWR-6 presents the variations in the wall temperature, heat flux and heat transfer coefficient with respect to the fluid temperature. The wall temperature gradually increases with fluid temperature below and above the pseudo-critical point. At a pseudo-critical point (384.9°C), a sharp jump appears in the wall temperature, in which it increases from 253.5°C to 317.4°C with an increase in fluid temperature from 375.7°C to 383°C. The heat flux increases with fluid temperature below the pseudo-critical point, and is almost maintained at a constant value above the pseudo-critical point, but decreases sharply near the pseudo-critical point itself. The heat transfer coefficient increases and then decreases with fluid temperature, reaching its maximum at 14 kW·m−2·K at the pseudo-critical point.

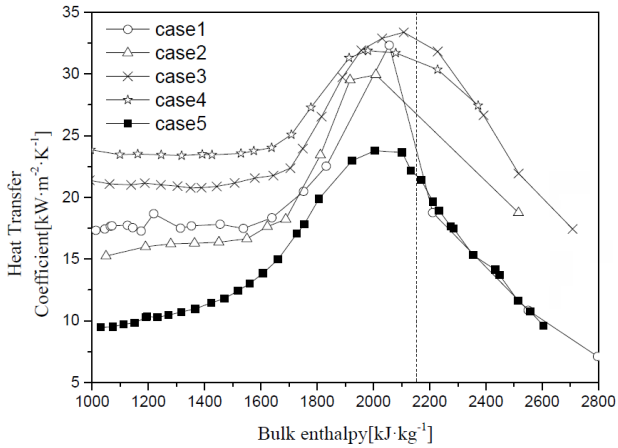
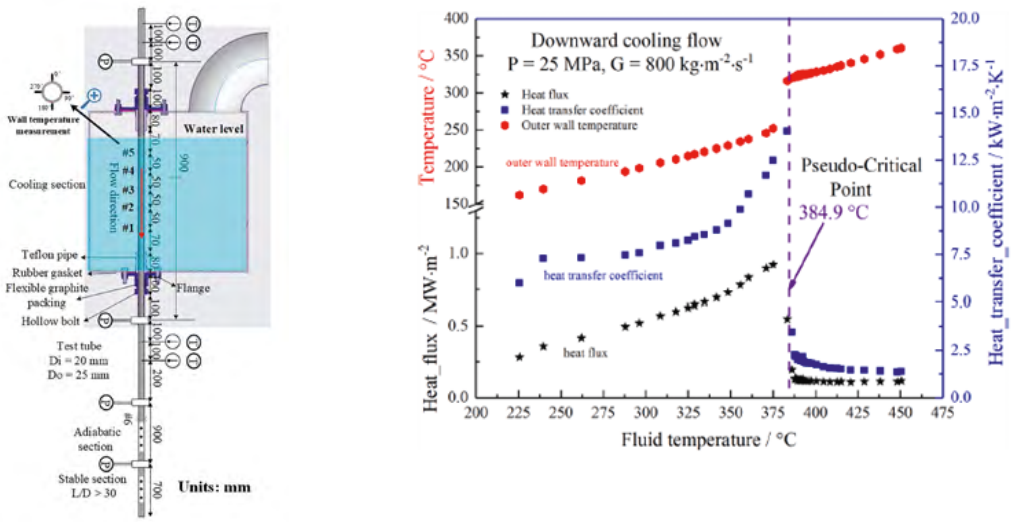


Figure SCWR-5. The distribution curve of heat transfer coefficient under supercritical conditions

Activities in Canada

In Canada, a preliminary SCW-SMR concept has been established from the reference Canadian SCWR concept. The overall core diameter is about 3 meters, which has resulted in a slender reactor core, inefficient from both neutronics and thermal hydraulics points of view. The selection and optimization of the SCW-SMR fuel assembly uses knowledge gained from the development of Canadian fuel bundles. It considers geometrical features (such as heated perimeter, flow area, sub-channel size) to minimize the maximum cladding temperature (MCT). From a thermal-hydraulics and reactor physics coupled analysis, the study focused on computing the reactor power distribution and the maximum channel power. Based on these results, a thermal-hydraulics analysis using the subchannel code ASSERT-PV was performed. The analysis focused on the maximum cladding temperature. Several concepts have been proposed, however, based on the results, and two concepts are being further investigated, namely: 1) THE CANFLEX-20 fuel bundle; and 2) the 64-element concept used for the Canadian SCWR concept. The assessment of these fuel bundles is ongoing and the results are still preliminary. Moreover, a proper and complete development requires linking to operational and

Figure SCWR-6. The downward flow test section (left) and the general behaviors (right)



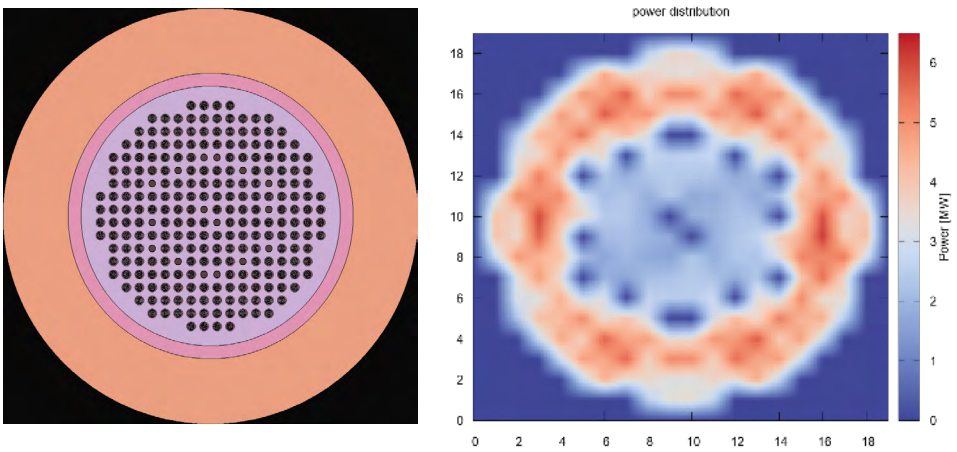


Figure SCWR-7. (a) Preliminary Canadian supercritical water-cooled SMR (core layout), (b) channel power heat map (in MW) of the preliminary Canadian supercritical water-cooled SMR

safety constraints and requirements, such as higher burn-up, maximum linear power, and higher-reliability or reduced operational and maintenance costs. These analyses are ongoing.

The CNL gravity assisted loop uses a heat pipe to remove heat from a pool of water. The experimental loop was originally designed to be used to remove heat from the spent-fuel pools. A study to examine the detailed behavior of this gravity assisted loop was recently undertaken. The loop was verified and tested to ensure it is fully operational, and a series of tests were conducted to provide experimental data that can be compared to a computer model. The objective was to further analyze the loop and to dimension components that could be used to design and assess a passive cooling system that could be included in the Canadian SCW-SMR. The system was modelled using CATHENA and RELAP5-3D codes. The investigation revealed unexpected effects in the steam piping between the evaporator and condenser due to the use of a Coriolis mass flow metre, which indirectly limited heat transfer. Additionally, the presence of non-condensable gases in the system further reduced heat transfer. The mathematical models were refined to include these factors, resulting in good agreement with experimental data. Further, it was demonstrated that changes to the loop, such as increasing the diameter of the condensate return line and relocating the mass flow metre are needed. Simulation results also provide guidance for the next phase of investigation, including the addition of a steam-to-air heat exchanger.

Materials and chemistry

Euratom activities

A European Union funded project called Mitigating Environmentally-Assisted Cracking Through Optimisation of Surface Condition (MEACTOS) is ongoing to study the effect of surface finishing on the corrosion resistance of selected alloys (A182 and stainless steel 316 L). In this project, one task involves SCW being used as an aggressive environment because of the higher test temperatures. Several laboratories from the EU are involved in these SCW activities (e.g. JRC, Valtion Teknillinen Tutkimuskeskus [VTT] and CIEMAT).

In the context of the ECC-SMART project, work package 2 is focusing on materials testing with more than 400 person months and the participation of 11 laboratories, among which 8 European laboratories/companies[1], one from Canada (CNL) and two from China (University of Science and Technology Beijing [USTB] and Shanghai Jiao Tong University [SJTU]). In this work package, the corrosion behavior in SCW of two of the most promising materials for cladding applications (i.e. 800 H and 310 S) will be studied.

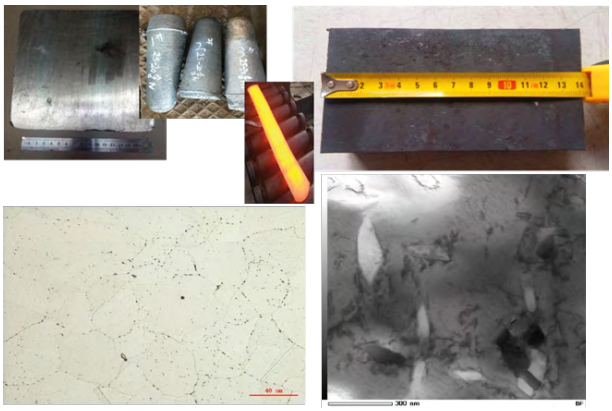
JRC Petten, in collaboration with VSCHT, has published a paper in the Corrosion Science Journal entitled: "In-situ electro-chemical impedance measurement of corroding stainless steel in high subcritical and supercritical water". This work summarized the main findings from a study on changes in the physico-chemical properties of SCW with pressure and temperature. This work could be considered the starting point of the tests to be carried out in the ECC-SMART project on this topic.

CIEMAT finished a preliminary work in collaboration with the Research Centre Řež. Selected, in-situ tensile tests with a nickel-based alloy 690 pretested in SCW were performed in this work. Both laboratories plan to continue these tests throughout 2021, and for this reason CIEMAT has designed and machined new specimens that fit into the scanning electron microscope available at the Research Centre Řež Pilsen, where the previous tests were performed. It is expected that this new configuration of specimens will allow a more in-depth study of the role of microstructural defects in the corrosion behavior of Ni base alloys. Preliminary results from these tests were presented at the Electric Power Research Institute (EPRI) annual meeting on alloy 690 (Tampa, December 2019).

1. CIEMAT, Research Centre Řež, JRC, Regia Autonoma Tehnologii Pentru Energia Nucleara [RATEN], Slovak University of Technology in Bratislava (STU), VTT, the University of Prague (VSCHT), ENEN (European Nuclear Education Network)

Research activities at the Research Centre Řež are linked to previous years, focusing on two more exposures in the SCWL, commissioning and test operation of the ultracritical water loop (UCWL) and publishing of data from exposures in 2018-2020. Effects of supercritical water on the corrosion behavior was studied on perspective materials for nuclear power plants, such as alloy 800H, AISI 310S, AISI 321SS (08Cr18Ni10Ti), T505 (T91), Inconel 718, NIMONIC 91 and NITRONIC 60. Two exposures in the SCWL were carried out at 395°C, 25 MPa, each for 1 000 hours. The supercritical medium was deoxygenated water with pH 5.5-6.8, conductivity 0.77-1.88 µS/cm, Fe <92 µg/l and TOC 886 to 251 µg/l. Commissioning of the UCWL took place during the year 2020 and the test operation at 600°C, 25 MPa, with 500 hours running since November 2020. The effects of SCW were evaluated by weight changes, scanning electron microscopy (SEM) with chemical analysis detectors (BSE, EDS, EBSD) in combination with X-ray diffraction. In addition, the special analysis to detect the very thin surface layer by focused ion beam (FIB) was used on alloy 800H. In this case, no thin surface layer was observed. Only randomly distributed spinels were observed: trevorite ($NiFe_2O_4$, a = 8.347 Å), magnetite ($FeFe_2O_4$, a = 8.397 Å) and chromite ($FeCr_2O_4$, a = 8.376 Å) crystals with dimensions up to 1 µm on 800H and only magnetite crystals on 08Cr18Ni10Ti after three exposures – 1 700 hours. The density of all crystals increased slightly after the second and third exposure. Simultaneously, the surface of Inconel was irregularly covered by oxide crystals about 1 µm, identified as hercynite and chromite $Fe(Al,Cr)_2O_4$ after one exposure – 1 000 hours. Other materials will be analyzed in 2021. Selected materials, such as 800H, AISI 321SS and T91, were exposed several times, for 2 700 hours (four exposures) in total. The commissioning of new facilities such as in-pile and out-of-pile autoclaves (volume 137 ml, 600°C/25 MPa and volume 850 ml, 700°C/30 MPa) are in progress. The effect of radiation on microstructure stability and corrosion resistance of candidate materials exposed to SCW will thus be possible to assess.

Figure SCWR-8. AFA steels developed at the University of Science and Technology Beijing (China)



Activities in China

The SJTU is studying the corrosion behavior of alloy 800H, austenitic stainless steel 310S and alumina forming austenitic (AFA) alloys in SCW. Moreover, particular interest is being taken in the effects of variables such as temperature, plastic deformation, water chemistry and surface finishing in the corrosion behavior of these alloys, and for this reason the SJTU is performing tests to study the effects of these variables on general corrosion and stress corrosion cracking processes in SCW. In addition, the SJTU is leading the third international round robin on corrosion behavior of candidate alloys for the SCWR. In this case, the international group will focus their efforts on the study of stress corrosion cracking processes in the SCW. Part of these activities will be complementary to the ECC-SMART project.

The USTB is designing and fabricating new grade materials suitable for fuel-cladding application under a high-temperature SCWR environment. These materials will be co-evaluated by colleagues at SJTU and the NPIC to determine the candidate materials for further round robin tests. The composition design of new grade materials is mainly based on 310SS, including oxide dispersion-strengthened (ODS) steel type fabricated through the powder metallurgy technique and AFA steel type through a traditional melting and casting method. Both types of new grade materials show promising high-temperature strength and SCW corrosion resistance. ODS steels show much better microstructure stability at high temperature, while AFA steels are attractive in terms of engineering and manufacturing. AFA steels show good performances in the aspect of high-temperature strength as compared with similar traditional steels because of the formation of strengthening phases of NbC, Laves and B2-NiAl, which are more stable than M23C6 in traditional steels. It can be expected that the AFA steels will also show better corrosion resistance in SCW environments as a result of the formation of alumina surface oxide, which is dense, thin and stable, as approved by exposure tests in SCW. Figure SCWR-8 shows the pictures of ingots, microstructure and precipitates of a fabricated AFA steel.

Activities in Canada

The CNL has studied the corrosion behavior in SCW of alloy 625 and alloy 800H. Both have showed an excellent strength and ductility after welding. In addition, Cr-coated Zr-2.5Nb, Zr-1.2Cr-0.1Fe, Ti and Ti-6Al-4V met performance criteria in short-term tests. Moreover, they have developed a schedule for proton and heavy ion (Cr3+) irradiation up to 5-15 dpa in top 20-30 µm, followed by micro-mechanical testing. It is expected that the irradiation will start during the year 2021.

Engineering, structural and core metallic nuclear components of NPPs must handle thermal loading; otherwise, localized hotspots resulting from changes in geometry or heat transfer fouling

could develop within the column and degrade the performance of components during operation and over time, such as that of fuel-cladding tubes. To further progress in this area, a project was established at the CNL with the objective of determining the thermal properties data (thermal conductivity) of candidate cladding tube material of small modular SCWRs, and to assist thermal-hydraulics calculations.

Chromium coated Zr-2.5% Nb material, such as a cladding material, showed a better resistance to corrosion in supercritical water conditions. In the open literature, empirical or semi-empirical models are available to assess the thermal conductivity of zirconium based fuel-cladding materials, but there is limited or no data on thermal conductivity of Cr-coated and/or oxided zirconium based fuel-cladding materials is lacking or not available to support thermal hydraulics modelling. Thermal conductivity measurements using the laser flash method were performed on the as-received material (Zr-2.5% Nb) at different test temperatures. The specimens are disc shaped with 12.16 mm in diameter, and an average thickness of 1.364 mm was cut from Zr-2.5% Nb. At each temperature, measurements were repeated five times to obtain an average. In the current study, thermal diffusivity and conductivity of baseline material is determined.

The CNL has an ongoing R&D program to support the development of a scaled-down 300 MWe version of the Canadian supercritical water reactor (SCWR) concept. The 300 MWe and 170-channel reactor core concept uses LEU fuel and features a maximum cladding temperature of 500°C (McLellan et al., 2021). There are challenges to using zirconium alloys at temperatures exceeding 400°C. Zirconium alloys such as Zr-2 and Zr-4 typically experience high corrosion rates, and they are known to experience hydrogen embrittlement from aggressive hydrogen pickup during corrosion.

Two materials from the previous experimental campaign – Zr-1.2Cr-0.1Fe (R60804) and Zr-2.5Nb (R60901) –were also used in the campaign described here. A nominal thickness of 5 to 10 μm of chromium coating was tested for about 150 hours of exposure time in oxygenated SCW. The oxidizing environment was chosen to simulate water radiolysis in the SCWR core. In addition, the corrosion behavior of candidate materials in an alkaline environment using LiOH solution was also evaluated.

Microstructural analyses, including scanning electron microscopy (SEM) and energy-dispersive X-ray spectroscopy (EDX) were performed to observe the effects of the microstructure of the base alloys on the observed chromium coating. The results from short-term autoclave oxidation at supercritical water conditions show that when the coating reaches approximately 10 μm thickness, the grain orientation of base Zr- and Ti-based alloys does not affect the morphology of the chromium coating. Moreover, weight gain measurements indicate a significant improvement in corrosion resistance of coated coupons compared to the as-received alloys, for Zr-2.5Nb and Zr-1.2Cr-0.1Fe. Long exposure experiments are ongoing.

Yanping Huang
Chair of the SCWR SSC, with contributions from SCWR members