25 YEARS OF SUPERCRITICAL WATER-COOLED REACTOR RESEARCH IN EUROPE: LESSONS LEARNED AND FUTURE CHALLENGES

Thomas Schulenberg
Karlsruhe Institute of Technology

Abstract

Research on Supercritical Water-Cooled Reactors started in Europe in 2000 with the European project High Performance Light Water Reactor, primarily to prepare codes for design and analyses of this new technology and to validate them. Furthermore, the design requirements could be specified. In its second phase, from 2006 and 2010, a nuclear reactor and power plant concept was worked out in detail, giving more insight into the technical challenges and providing a first estimate of the economic advantages of such a power plant of the 4th generation. Some open issues, which remained from this project, are being addressed today in a design study of a small modular reactor concept, to be cooled with supercritical water. In the meantime, an in-pile fuel qualification test under supercritical water conditions was designed and prepared for licensing, as a first step towards a real prototype reactor. Up to now, however, neither this in-pile test nor a prototype reactor are expected to be built in near future. Nevertheless, these international projects could provide a valuable contribution to nuclear education and training on water-cooled reactor design.

Keywords: Supercritical water-cooled reactor, power plants, design, review, European projects

1. Introduction

Following some early attempts to use supercritical water as coolant for light water reactors, an initiative to study such concepts systematically again was taken by Y. Oka in the late 1990s. Together with some students from the University of Tokyo, he worked out the first core design concepts with square or hexagonal fuel assemblies, which were equipped with water rods to yield a thermal neutron spectrum, compensating the low coolant density at higher temperature, or without water rods for a fast neutron spectrum. A book by Oka et al. [1], issued later in 2010, is still providing the best overview of their studies. Their target design data, i.e. a feedwater temperature of 280°C and a core outlet temperature of 500 °C at a pressure of 25 MPa, were taken later by many other design studies outside Japan as reference data. Moreover, they focused already on a once-through steam cycle, such that the superheated steam produced inside the core was considered to be supplied directly to a high pressure steam turbine. “Once-through” means that reactor coolant pumps are not required, and just the feedwater pumps of the steam cycle provide the coolant mass flow through the reactor core. Consequently, this concept was cheaper than any Pressurized Water Reactor (PWR), with its steam generators and pumps, or Boiling Water Reactor (BWR), with its steam separators, driers and recirculation pumps. As the plant efficiency and power per mass flow rate were significantly higher, on the other hand, this Supercritical Water-Cooled Reactor (SCWR) concept was expected to become more economic than any current light water reactor.

These obvious advantages drew also the attention of vendors of nuclear power plants in Europe and in the United States. In Europe, the former nuclear departments of Siemens, in particular, who became Framatome ANP, Germany, were interested to find out if their BWR concepts could be improved such that they became superior to any PWR, giving them a better position within the new French group.

Around 2000, this coincided with the urgent need to recruit a new generation of nuclear engineers, as their first generation was going to be retired soon. Designing a nuclear power plant from scratch was a challenging target motivating students to study nuclear engineering, not only in Europe, and the SCWR was close enough to current light water reactor technologies to prepare them for a later job in the nuclear industry.

This reactor concept was further supported by the initiative of the Generation IV International Forum (GIF), who selected the SCWR as one of the six most promising reactor concepts to be studied jointly in more detail. It took until 2003, however, until Euratom signed the GIF Charta, enabling all their member states to participate at the world-wide collaboration.

2. High Performance Light Water Reactor

With this background, a first European project on SCWR concepts started in 2000, coordinated by D. Squarer and financially supported by the European Commission in their 5th Framework Programme. Framatome ANP and Electricité de France (EdF) participated as industry partners, as well as the major European research organizations. The consortium succeeded to get also Prof. Y. Oka on bord, enabling them to use results of the University of Tokyo as reference cases. Trying to avoid the term "supercritical" in the title of the project, which could be misunderstood neutronically, they called their reactor concept the High Performance Light Water Reactor (HPLWR) [2].

Taking the hexagonal core design concept of Dobashi et al. [3] and the square core design concept of Yamaji et al. [4] as reference cases, the focus of the European project was rather to check the available neutronic and thermal-hydraulic codes for their applicability to supercritical water and to prepare sub-channel codes and system codes for the planned design work. The envisaged reactor concept with a conventional single pass core is sketched in Fig. 1. Additional moderator water inside the water rods of the fuel assemblies was considered to run downwards, and supercritical water as coolant was to run only upwards through the core, being heated up from 280 °C at the core bottom to 508 °C at the core top, at a system pressure of 25 MPa. The fuel rods had an outer diameter of 8 mm and the active core height of 4.2 m was similar as with current PWR design. Control rods were planned to be inserted from top like in a PWR. The overall plant concept featured a containment and a steam cycle like the latest BWR design of Framatome ANP, but designed for a higher pressure. The gross plant output was set at 1000 MW, such that a steam cycle with a 50 Hz steam turbine, as used for recent coal fired power plants, could serve as a basis. The thermal efficiency of such a cycle is approximately 44%, indicating already a clear economic advantage compared with other light water reactors.

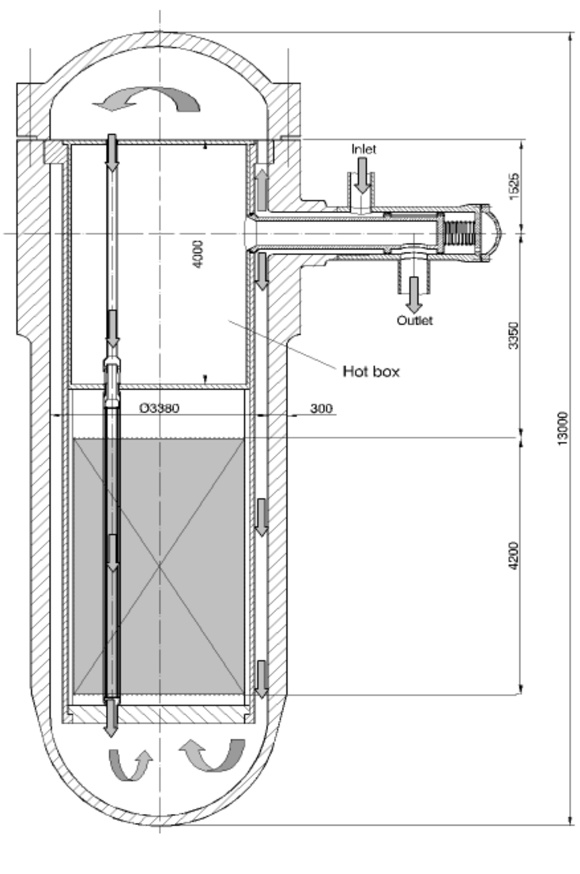


Fig. 1: Reactor pressure vessel design of the HPLWR, phase 1 [2]

The HPLWR project included also a study of candidate materials for fuel claddings, pressure vessel and pipings [5]. While ferritic steel like 20MnMoNi 5 5, cladded with stainless steel on the inside as used for current PWRs, can still be taken for the HPLWR pressure vessel, the fuel claddings will be exposed to higher temperatures. Zircalloy can certainly not be used anymore. Instead, austenitic stainless steels like 1.4970 were considered as a good compromise with sufficient creep strength up to 650 °C, compared with Ni-base alloys with better creep and corrosions resistance, but which are causing more neutron absorption and embrittlement. Corrosion data under these supercritical water conditions, however, were rather limited at this time.

Almost in parallel, just with one year delay, a feasibility study on SCWRs was performed in the United States by INEEL and Westinghouse [6]. Similar as in the European project, they took the core design concept of Yamaji et al. [4] as a starting point for their own design concept, featuring a single pass core with 280 °C at core inlet and 500 °C at 25 MPa at core outlet, but designed for a net electric power of 1600 MW of the steam cycle. As candidate cladding material for fuel rods, they considered stainless steel 316L, Alloy 690 or MA956, a ferritic ODS alloy. These target data were quite comparable to the European study, but their analyses of the core were far more detailed, including also studies of non-uniform power profiles and hot channels inside the core. Surprisingly, their hottest fuel rods exceeded all reasonable material temperature limits, and the study ends with the conclusion that a SCWR with the reference core design does not appear to be feasible. This result is due to the fact that the specific heat of steam at 600°C and 25 MPa is 3 times smaller than the average specific heat from core inlet to outlet, causing a steep increase of coolant temperatures with any additional heat or reduced mass flow inside hot channels. The SCWR is thus far more sensitive to hot channel conditions than a PWR, where the coolant temperature is limited by the saturation temperature.

While the US consortium gave up the SCWR at this point, the European consortium took the challenge and modified their overall design strategy accordingly, such that the coolant heat-up inside the core is always limited by the hottest fuel rod. As long as the average coolant temperature is lower than the envisaged core outlet temperature, the coolant must be mixed to get rid of the hot streaks, before it may further be heated up. Consequently, as coolant mixing inside the core is obviously impossible, this strategy leads to a multi-pass core with coolant mixing above or underneath the core. The smaller the difference between the maximum allowable peak coolant temperature and the envisaged average core outlet temperature, the more mixing stages will be required [7].

3. HPLWR Phase 2

Having understood the basic design requirements, the HPLWR was designed and analysed in detail in the second phase of the project from 2006 to 2010, as part of the 6th Framework Programme. The consortium was similar as in the first phase, except that EdF did not participate anymore, but other research organizations entered the consortium instead. The project was managed by J. Starflinger. Results have been documented in [8], where also the large number of students and young scientists is listed who contributed to this project with their Master or PhD thesis. Indeed, the contribution of this project to nuclear education and training can be seen as one of the major achievements of this project.

Aiming at a net electric power of 1000 MW, the reactor core should produce a thermal power of 2300 MW at a net thermal efficiency of 43.5 %. Constraints for core design were a peak cladding surface temperature of 630 °C and a peak coolant temperature of 600 °C, at an average core outlet temperature of 500 °C.

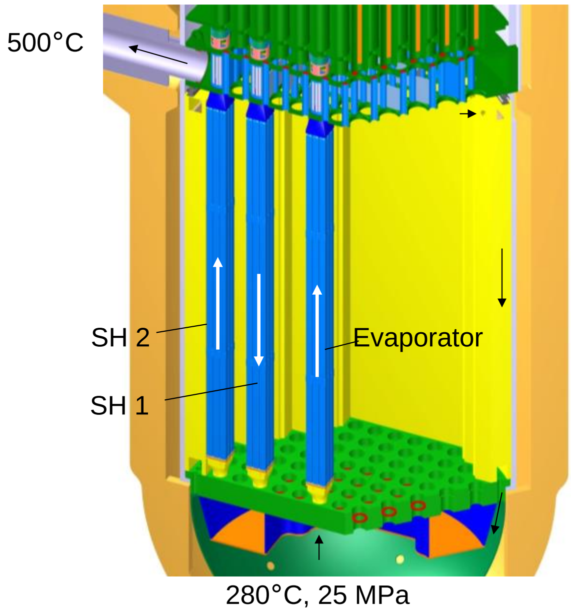


Fig. 2: Three-pass core design of the HPLWR, phase 2 [8].

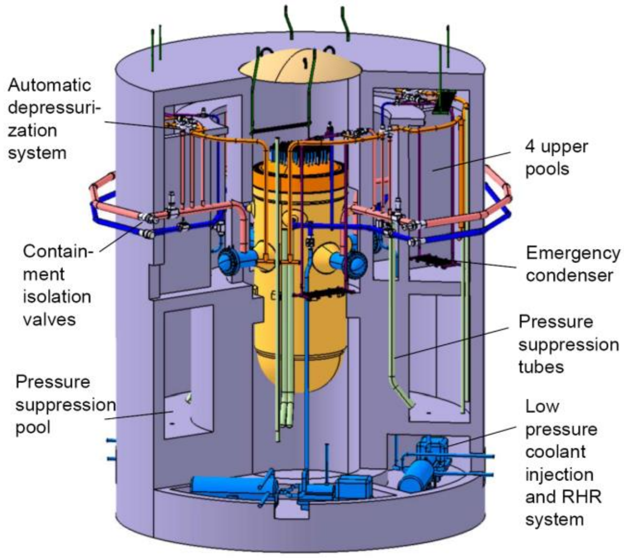


Fig. 3: Containment design of the HPLWR, phase 2 [8]

3.1 Core design

The reactor core was designed with three heat-up steps with coolant mixing above and underneath the core, as shown in Fig. 2. The coolant was entering the reactor pressure vessel with a temperature of 280 °C and a pressure of 25 MPa. Half of the total mass flow of 1180 kg/s was first used as moderator water, which was guided through water rods inside the fuel assemblies, through the gaps between the assembly boxes and through the reflector surrounding the core, to be mixed underneath the core with the other half through the downcomer. Together, it was heated up in a first step inside the 52 central assembly clusters to around 400 °C, which were called the evaporator assemblies (EVA). After being mixed in an inner plenum above the core, the coolant was driven downwards through another 52 assembly clusters surrounding the evaporator, providing the first superheater (SH1). At their outlets, the coolant was producing a swirl flow inside an annular mixing plenum underneath the core, before it was finally heated up to 500 °C in the outer assembly clusters at the core periphery, called the second superheater (SH2). The foot piece of such an assembly cluster is shown in Fig. 4a). It combines 9 assemblies with 40 fuel rods each and with wire wraps as spacers, Fig. 4b), of which 5 assemblies can be equipped with control rods running inside their central water box. The arrangement of these clusters as evaporator and superheater regions is shown in Fig. 4c).

A large number of analyses have been performed for this concept, as documented in [8], including fuel shuffling strategies and associated burn-up analyses, control rod effects, effects of burnable poisons etc., each even down to the details of single fuel rods. If local effects, uncertainties and allowances for operation are included, it was not easy to stay within a hot channel factor of 2, which had rather been considered as a conservative estimate at the beginning of the project. The hot channel factors at the beginning and end of an equilibrium burn-up cycle (BOC, EOC) and the coolant enthalpies, in average and those of the hot channels, are shown in Fig. 5. The target peak coolant enthalpy of 3500 kJ/kg, corresponding with a coolant temperature of 600 °C at 25 MPa, was still exceeded in some cases. With some further optimization of fuel shuffling, however, it was expected that a maximum cladding temperature of 650°C should be feasible. These studies demonstrated, however, that optimization of such a core requires more effort than for a conventional PWR design.

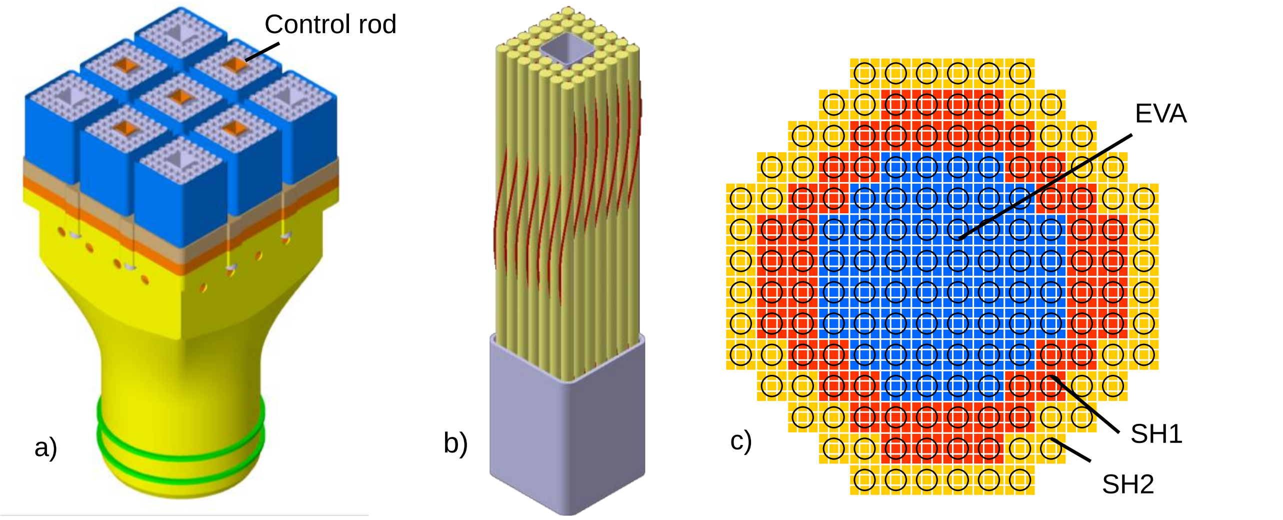


Fig. 4: Details of the reactor core of the HPLWR, phase 2: a) foot piece of an assembly cluster, b) single assembly with 40 fuel rods and wire wrap spacers, c) arrangement of fuel assembly clusters as evaporator (EVA), superheater 1 (SH1) and superheater 2 (SH2) assemblies [8].

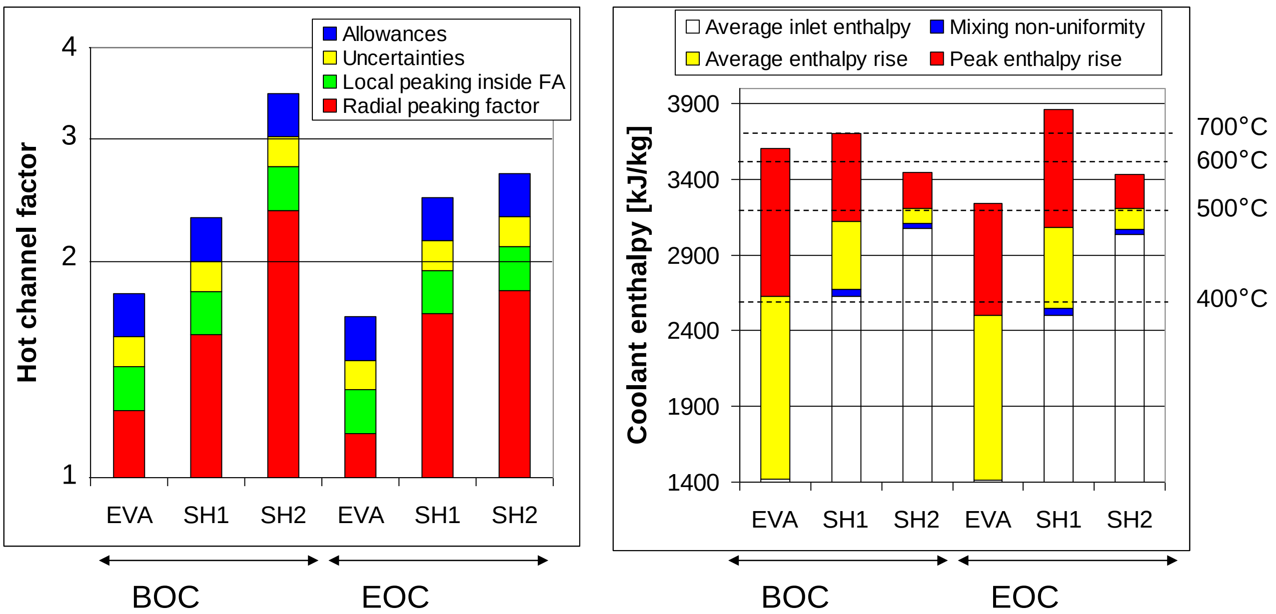


Fig. 5: Hot channel factors and coolant enthalpies at beginning and end of an equilibrium burn-up cycle (BOC, EOC) for each heat-up step of the HPLWR core design, phase 2 [8].

Some candidate materials for fuel claddings were tested in autoclaves at JRC and at VTT as part of this project. At supercritical pressure and at the maximum peak cladding temperature of 650 °C, the 600 hour tests gave a first impression of the corrosion resistance of stainless steel, ODS and nickel base alloys [9]. Results of the measured oxide thickness are plotted in Fig. 6 vs. the Chromium content of these alloys. Stainless steels like 316NG and 1.4970 produced already an oxide thickness of 100 µm, about as much as ferritic-martensitic steels like P91 and P92. At least 20% Chromium is needed for acceptable corrosion resistance, for which the ODS alloy PM2000 and Incoloy 800H are typical candidates.

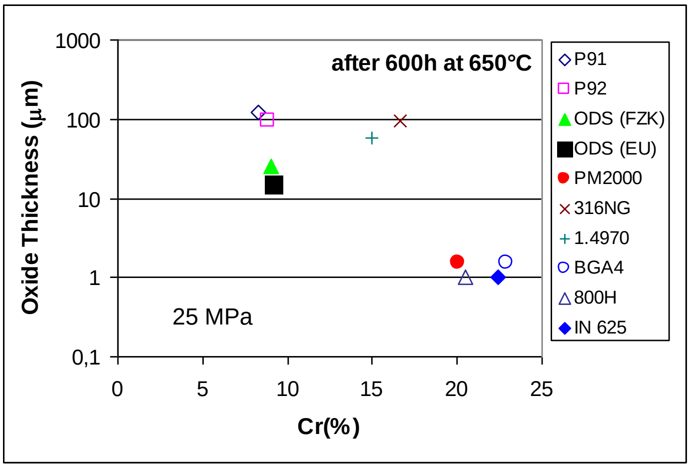


Fig. 6: Results of autoclave corrosion tests at 25 MPa at JRC and VTT [8].

3.2 Safety systems

The chosen containment design, Fig 3, shows the typical basic features of Framatome's BWR design at that time. As outlined by Schulenberg and Visser [10], the containment must include:

- a reactor shut down system by control rods or by a boron injection system as a second, divers shut down system;
- active and passive containment isolation valves to close the third barrier in case of an accident;
- steam pressure limitation by pressure relief valves;
- automatic depressurization of the steam lines into the upper pools inside the containment through spargers, to close the coolant loop inside the containment in case of containment isolation;
- a coolant injection system to refill coolant into the pressure vessel after intended or accidental coolant release into the containment;
- a pressure suppression pool to limit the pressure inside the containment in case of steam release inside the containment;
- a residual heat removal system for long-term cooling of the reactor core, which could be either driven by active pumps in the basement of the containment or by an emergency condenser inside the upper pool, together with passive containment condensers.

With an inner height of just 23 m, less than half the size of a BWR containment, the HPLWR containment appeared to be surprisingly compact. Several system code analyses, which were performed with APROS, CATHARE, SMABRE and RELAP5 [8] for loss of flow, loss of coolant, and loss of power accidents confirmed indeed that this containment was large enough to prevent a severe core damage. A challenge appeared to be a break of a feedwater line, causing the highest cladding surface temperatures. Therefore, a backflow limiter at the inlet nozzles was foreseen to minimize coolant losses through the feedwater line.

While an active low pressure coolant injection system in the basement of the containment could certainly remove the residual heat over long term, the passive residual heat removal system via the emergency condensers would not work simply driven by gravity. Steam trapped inside the mixing plenum above the core could hinder a natural convection flow, so that either a steam turbine driven pump or a steam injector would be required to drive the recirculating flow [10].

3.3 Steam cycle

The steam cycle designed for this power plant, Fig. 7, was based on turbine technologies for coal fired power plants. At a net electric power of 1000 MW, the high pressure (HP) / intermediate pressure (IP) / low pressure (LP) turbine train could run at 50 Hz, reducing the turbine mass by around 50 % compared with half speed turbines of a conventional BWR of similar power. Two parallel lines of feedwater systems, with three low pressure preheaters and four high pressure preheaters each, were based as well on fossil fired power plant technology. Four parallel feedwater pumps were foreseen, of which three pumps can provide the full mass flow rate and one was kept on hot stand-by to continue operation in case of a pump trip. Different from fossil fired power plant technology were the two parallel reheaters between the HP and IP turbine, which were designed as straight tube heat exchangers and which did not require a moisture separator.

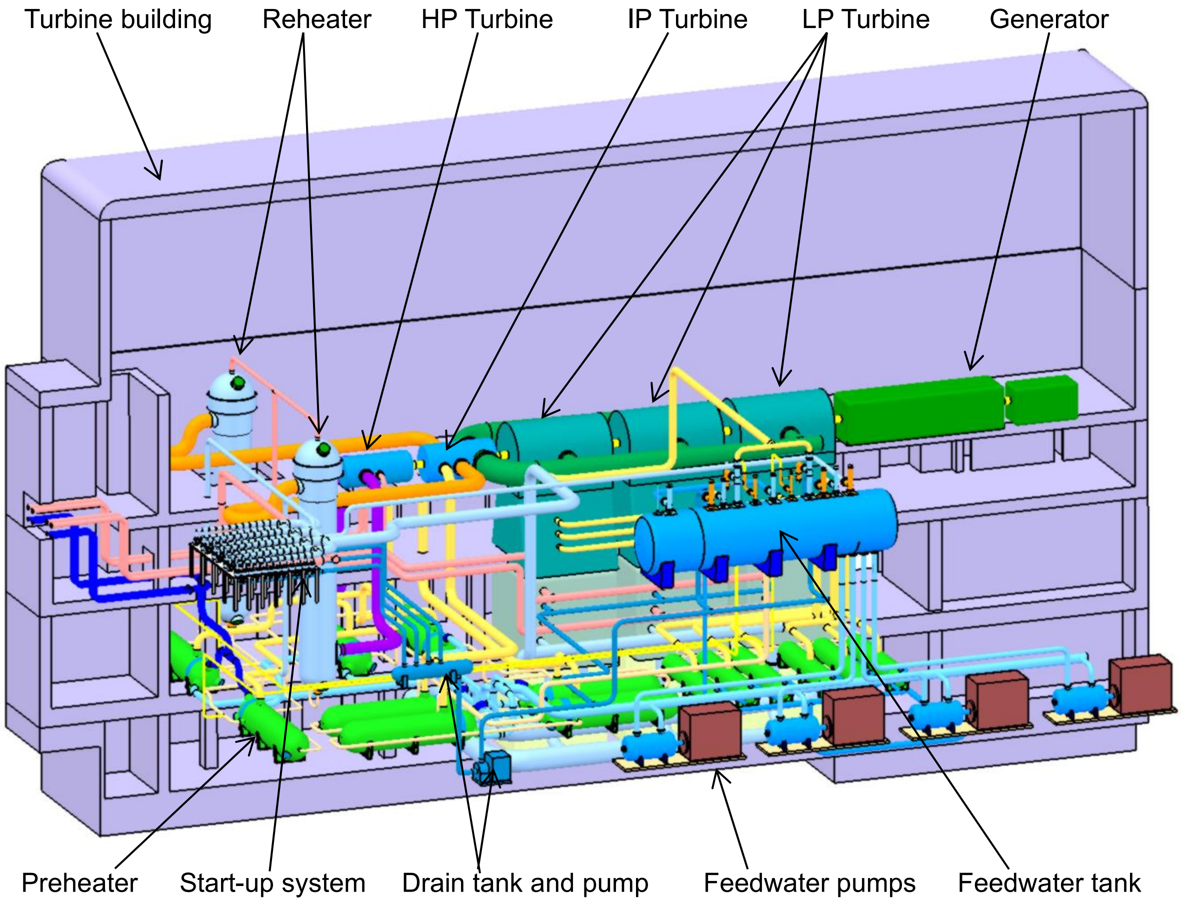


Fig. 7: HPLWR turbine building with steam cycle components

Quite different from fossil fired power plants was also the start-up procedure. A coal fired power plant is usually operated with a sliding pressure, such that the system pressure increases proportionally with load. Consequently, the steam generator is operated at supercritical pressure only in the upper load range and is running at subcritical pressure otherwise. Such operation is causing dryout of the boiler tubes, and their surface heat flux had to be limited to 400 kW/m^2 to avoid material damages under post-dryout conditions. The heat flux of fuel claddings of the HPLWR, however, can reach up to 1550 kW/m^2, so that any operation at subcritical pressure must definitively be avoided, except for residual heat removal. Therefore, the turbine governor valves had to keep the reactor pressure vessel at supercritical pressure in the entire load range. Below 50 % load, when the feedwater pumps were running at their minimum flow rate, the steam enthalpy was too low to be supplied to the turbine, and the hot water coming from the reactor pressure vessel had to bypass the turbine and to be expanded in a battery of cyclone separators instead [8]. It is indicated as the start-up system in Fig. 7.

3.4 Economic assessment

Thanks to the collaboration with Framatome in the HPLWR project, the plant erection costs could be estimated quite realistically. Despite the huge savings in the containment and in turbine size, the total savings of the HPLWR overnight cost were just about 25 % compared with an advanced BWR [8]. The higher net efficiency would almost balance the higher fuel costs, for which a higher enrichment and more structural material would be needed than for a BWR. Taking also the additional costs into account, which must be paid for research and development, the costs of a prototype and for the first of a kind, which must be paid back by the fleet later on, there was hardly any economic benefit left. In the meantime, Framatome gave up the BWR concept and consequently withdrew from further SCWR development. The HPLWR project concluded from this exercise that a continuous development from an existing PWR or BWR in an evolutionary process might be more economic.

4. ECC-SMART

Two remaining technical issues of the HPLWR concept, namely the hot peak cladding temperatures and the need of a pump for residual heat removal could be solved with an even more sophisticated core design, which is being studied in a recent project ECC-SMART, aiming for a small modular reactor with passive safety systems. As proposed by Schulenberg and Otic [11], a core design with horizontal fuel assemblies and seven heat-up steps, Fig. 8, each with intermediate coolant mixing, could further reduce the peak cladding temperatures and enable a gravity driven residual heat removal system, as the coolant flow is only horizontal or upwards inside the core. Results of some first analyses performed for this ambitious concept will be shown at the ISSCWR-11 symposium in 2025 to assess its viability. But the effort to fully design, test and build such an innovative reactor would certainly be high.

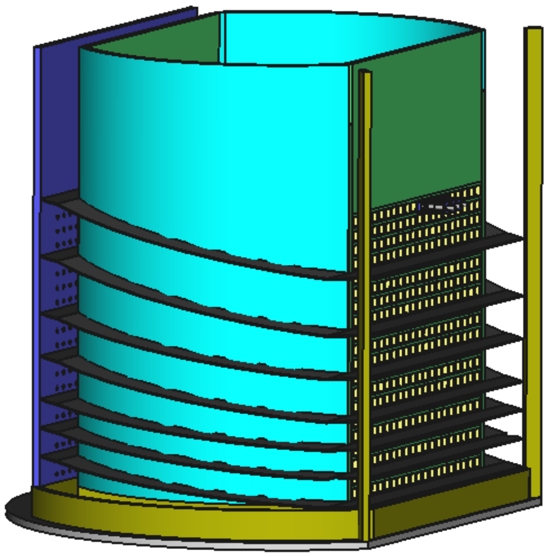


Fig. 8: Concept of a small modular reactor with horizontal fuel assemblies and seven heat-up steps with intermediate mixing [11].

5. Fuel Qualification Test

In Nov. 2006, the Joint Research Centre of Euratom, on behalf of the Euratom member states, signed a System Arrangement for international research and development of SCWR nuclear energy systems. In their System Research Plan, the SCWR steering committee proposed a prototype fuelled loop as a first and mandatory step before a demonstration unit of any future SCWR can be built. This in-pile test with prototype fuel rods and nuclear heating inside a supercritical water loop was designed, analysed and even tested out-of-pile in a joint international project SCWR Fuel Qualification Test (SCWR-FQT), which was funded by the European Commission in their 7th Framework Programme and performed jointly with 9 Chinese research organisations in their project SCRIPT. The project was coordinated by M. Ruzickova in 2011 to 2014, just following the HPLWR project, with the objective to insert a pressure tube with four fuel rods of 60 cm active length into the SVR-15 research reactor at CVR in Rez, Czech Republic, to be connected with auxiliary and safety systems next to the reactor building [12].

Besides the fuel rods with wire wraps, Fig. 9a), the active channel to be operated at 25 MPa included a recuperator, Fig. 9b), and a heater or cooler to produce a coolant temperature of 366 °C at the inlet of the test section. As a critical arrangement inside the reactor core, the fuel rods should produce 63.6 kW fissile power and 9.8 kW γ-power to heat up the coolant to 383 °C, just below the pseudo-critical temperature. In the sense of the HPLWR core design, this means that the test should simulate part of the evaporator. Two guide tubes around the test section should keep the temperature of the pressure tube below 350 °C.

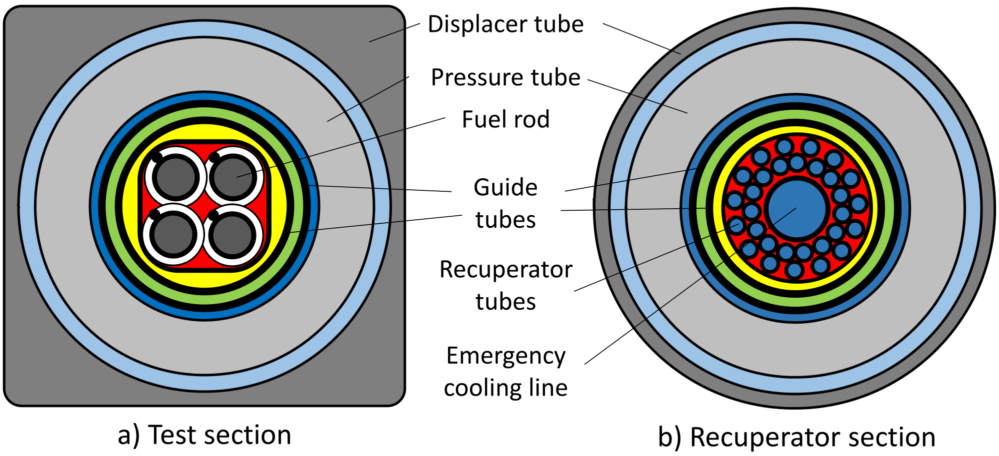


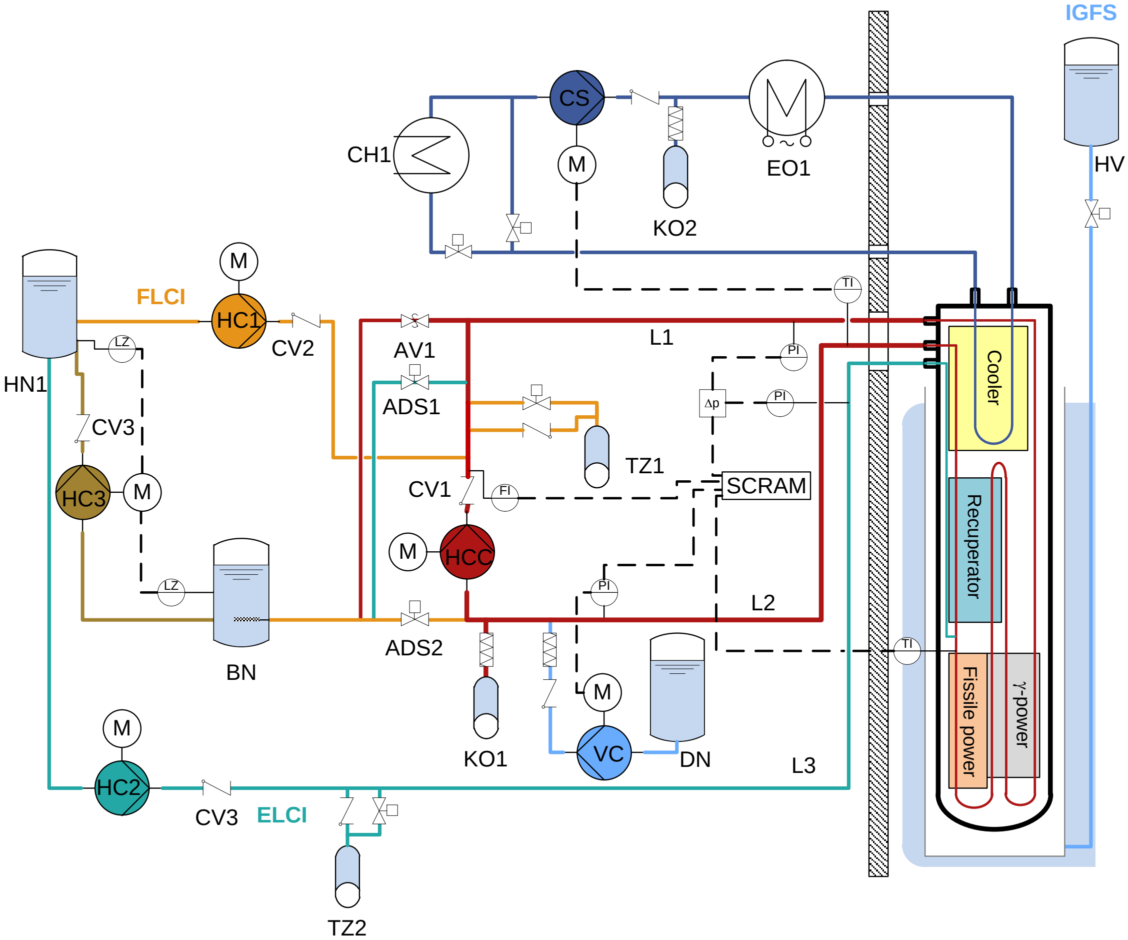
Fig. 9: Active channel with four fuel rods of the SCWR-FQT project [12]

An out-of-pile test of the test section under full pressure was performed by the Chinese partners in their project SCRIPT. With their SWAMUP test facility in Shanghai, they could not only measure the steady-state heat transfer of this small fuel bundle, but also the transient heat transfer during depressurization [13]. There we see that cladding surface temperatures may become a lot hotter as soon as the pressure falls below the critical pressure. This is important to know not only for the start-up or shut-down procedure, but also for accidental conditions without scram.

These tests, as well as detailed CFD-analyses of the test section, indicated a peak cladding surface temperature of less than 500 °C under normal operating conditions, so that stainless steel could be used as cladding material. Even stainless steel 316L, which is already qualified and available for nuclear applications, would be acceptable. Autoclave tests of these materials at 25 MPa were performed during the project at 500 °C and 550 °C, confirming sufficient corrosion resistance [12].

The auxiliary and safety system included a pressurizer, an accumulator and a recirculation pump, a pressure relief valve and an automatic depressurization system into a pressure suppression tank, as well as emergency cooling systems into the inlet and outlet of the active channel. The system, shown in Fig. 10, is thus similar to a PWR, but certainly a lot smaller.

The system has been analysed by Raqué [14] using the system code APROS, assuming normal operation as well as design basis accidents like: trip of the recirculation pump, loss of heat sink, loss of power, blockage of the coolant flow path, coolant shortcut inside the test fuel element, and loss of coolant accidents. Additionally, the mechanical and radiological consequences were studied for the case of an unlikely, but severe, multi-component failure beyond design basis accidents [15]. These analyses demonstrated that a secondary failure of the research reactor can always be excluded. They indicated, however, that the use of active fuel inside a high pressure test loop requires a closed containment around the loop and its supply systems to avoid radioactive release to the environment under worst case assumptions. The SCWR fuel qualification test was designed accordingly with a closed confinement around the entire test facility, which is strong enough to withstand even a postulated explosion of hydrogen, which might be produced during such postulated severe accidents. This project has certainly been an excellent exercise to prepare licensing documents for a nuclear facility operated with supercritical water, but CVR could not be convinced up to now to build and operate this test facility.



ADS – Automatic depress. system	DN – Refilling tank	HN1 – Emergency coolant reservoir
AV – Pressure relief valve	HC – Emergency pump	HV – Reservoir
BN – Pressure suppression tank	HC2 – Emergency pump 2	KO – Pressurizer
CV – Check valve	HC3 – Refilling pump	TZ – Pressure accumulator
EO1 – Electrical heater	HCC – Recirculation pump	VC – Boost pump

Fig. 10: Sketch of the SCWR-FQT loop and its safety systems [13].

6. Conclusions

Other design studies like the ones reported here have been performed in Japan, Canada and China. None of these concepts, however, is likely to be built in near future. The sad experience that not even the smallest version of a SCWR, the FQT facility, is going to be realized, which has been fully designed and documented as required for licensing and which would be financially affordable, is really disappointing. We learn from this experience that design and analyses are easily performed at a computer, and even out-of-pile experiments, which are more expensive, can be realized. As soon as we talk about radioactive tests, however, we experience a huge resistance, giving doubts that any of the visionary Generation IV concepts will ever be built. At least, the path to realization will be long.

The greatest achievement of all these studies is rather the large number of young nuclear experts, who learned all disciplines of nuclear engineering by solving challenging tasks of designing a nuclear reactor from scrach. Most of the students, who contributed to these projects, were emloyed later by the nuclear industry, by nuclear power plants, by technical inspectors and even by nuclear regulators. Several of them can still be found in nuclear research organisations. As a visionary dream, the SCWR is highly motivating to exceed the current limits of technology (the racing car among the water cooled reactors!), and these students succeeded to carry over their motivation to conventional PWRs or BWRs. Quite important was the close cooperation with the nuclear industry during these projects, giving the students not only the experience of senior experts, but also the impression that their work is more than just academic research.

Unfortunately, these cooperations with the nuclear industry were significantly reduced in recent years, and research organizations, who are still studying the SCWR, are well adviced to approach them again. A study of an evolutionary development step from current PWRs towards an SCWR could bridge the gap between these visionary dreams and reality [16].

7. References

[1] Oka, Y., Koshizuka, S., Ishiwatari, Y., Yamaji, A. (Eds.), 2010. Super Light Water Reactors and Super Fast Reactors. Springer US, Boston, MA.

[2] Squarer, D., Schulenberg, T., Struwe, D., Oka, Y., Bittermann, D., Aksan, N., Maraczy, C., Kyrki-Rajamäki, R., Souyri, A., Dumaz, P., 2003. High performance light water reactor, Nuclear Engineering and Design 221, 167–180

[3] Dobashi, K., Oka, Y., Koshizuka, S., 1998. Conceptual design of a high temperature power reactor cooled and moderated by supercritical light water. In: Proceedings of the Sixth International Conference on Nuclear Engineering. ICONE6, ASME, NY.

[4] Yamaji, A., Oka, Y., Koshizuka, S., 2001. Conceptual core design of a 1000MWe supercritical pressure light water cooled and moderated reactor. ANS/HPS Student Conference Texas A&M University

[5] Ehrlich, K., Konys, J, Heikinheimo, L., 2004. Materials for high performance light water reactors, Journal of Nuclear Materials 327, 140–147

[6] McDonald, P., Buongiorno, J., Strebentz, J.W., Davis, C., Witt, R., Was, G., McKinley, J., Teysseyre, S., Oriani, L., Kucukboyaci, V., Conway, L., Jonsson, N., Liu, B., 2005. Feasibility study of supercritical light water cooled reactors for electric power production, INEEL/EXT-04-02530

[7] Schulenberg, T., Starflinger, J., Heinecke, J., 2008. Three pass core design proposal for a high performance light water reactor, Progress in Nuclear Energy 50, pp. 526-531

[8] Schulenberg, T., Starflinger, J., 2012. High performance light water reactor – design and analyses, KIT Scientific Publishing, Karlsruhe, ISBN 978-3-86644-817-9

[9] Pentillä, S., Toivonen, A., Heikinheimo, L., Novotny, R., 2006. Corrosion studies of candidate materials for European HPLWR, Proc. ICAPP ’08, Anaheim, CA USA, June 8-12, 2006, Paper 8163

[10] Schulenberg, T., Visser, D.C., 2013. Thermal-hydraulics and safety concepts of supercritical water cooled reactors, Nuclear Engineering and Design 264, 231-237

[11] Schulenberg, T., Otic, I., 2021. Suggestion for design of a small modular SCWR, 10th International Symposium on SCWRs (ISSCWR-10), Prague, the Czech Republic, March 15-19, 2021
[12] Ruzickova, M., Vojacek, A., Schulenberg, T., Visser, D. C., Novotny, R., Kiss, A., Maraczy, C., Toivonen, A. 2016. European Project "Supercritical Water Reactor-Fuel Qualification Test": Overview, Results, Lessons Learned, and Future Outlook. Journal of Nuclear Engineering and Radiation Science, 2(1), Article 011002
[13] Li, H.B., Zhao, M., Hu, Z.X., Gu, H.Y., Lu, D.H., 2017. Experimental study on transient heat transfer across critical pressure in 2x2 rod bundle with wire wraps, Int. Journal of Heat and Mass Transfer 11, 68-79
[14] Raqué, M., 2014. Safety analysis for a fuel qualification test with supercritical water, Dissertation, Karlsruhe Institute of Technology, KIT Scientific Reports 7682
[15] Schulenberg, T., Raqué, M., Zeiger, T., 2015, Expected safety performance of the SCWR Fuel Qualification Test, ISSCWR-7, 1033, 15-18 March 2015, Helsinki, Finland
[16] Schulenberg, T., Czifrus, S., 2025. Design concept of a Super Pressurized Water Reactor, 11th International Symposium on SCWRs (ISSCWR-11), P005, Pisa, Italy, February 3-5, 2025