European supercritical water cooled reactor

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A B S T R A C T

The High Performance Light Water Reactor (HPLWR), how the European Supercritical Water Cooled Reactor is called, is a pressure vessel type reactor operated with supercritical water at 25 MPa feedwater pressure and 500 °C core outlet temperature. It is designed and analyzed by a European consortium of 10 partners and 3 active supporters from 8 Euratom member states in the second phase of the HPLWR project. Most emphasis has been laid on a core with a thermal neutron spectrum, consisting of small fuel assemblies in boxes with 40 fuel pins each and a central water box to improve the neutron moderation despite the low coolant density. Peak cladding temperatures of the fuel rods have been minimized by heating up the coolant in three steps with intermediate coolant mixing. The containment design with its safety and residual heat removal systems is based on the latest boiling water reactor concept, but with different passive high pressure coolant injection systems to cause a forced convection through the core. The design concept of the steam cycle is indicating the envisaged efficiency increase to around 44%. Moreover, it provides the constraints to design the components of the balance of the plant. The project is accompanied by numerical studies of heat transfer of supercritical water in fuel assemblies and by material tests of candidate cladding alloys, performed by the consortium and supported by additional tests of the Joint Research Centre of the European Commission. Besides the scientific and technical progress, the HPLWR project turned out to be most successful in training the young generation of nuclear engineers in the technologies of light water reactors. More than 20 bachelor or master theses and more than 10 doctoral theses on HPLWR technologies have been submitted at partner organizations of this consortium since the start of this project.

1. Introduction

Following the trend of coal fired power plants in the last 20 years, the evolutionary development to higher temperatures and pressures, which are meanwhile exceeding even the critical pressure of water, has been considered by the Generation IV International Forum (GIF) as an option also for future light water reactors. The higher steam enthalpy could enable a direct, once through steam cycle such that neither steam generators nor steam separators and dryers would be required, and even primary coolant pumps could be omitted. Moreover, steam turbines and re-heaters could be significantly smaller than today, while the steam cycle efficiency would even be higher. As fossil fired power plants with supercritical steam conditions have been operated since 20 years now, reaching 600 °C life steam temperature or even more, this nuclear plant con-
cept can benefit from proven design of turbines, feedwater pumps and most other components of the steam cycle except the re heater. Moreover, the containment design can basically be derived from latest boiling water reactors, so that the research and development program is concentrating mainly on the reactor itself.

In Europe, a consortium of 10 organizations from 8 European countries, i.e., AREVA NP, CEAT, KFKI, NRG, PSI, CVR, VTT, and the Universities of Stockholm and Stuttgart, decided in 2006 to address this challenge by working out a design concept of such a reactor, which they called the High Performance Light Water Reactor (HPLWR), with a core exit temperature of at least 500 °C at a supercritical system pressure of around 25 MPa. The design objectives were a net electric power of 1000 MW and a net plant efficiency of around 44%, while the specific plant erection costs should not exceed 1000 €/kWe thanks to all these plant simplifications, considering the specific overnight capital costs of 1200 €/kWe for a Light Water Reactor (LWR) as a basis, as specified in the European Utility Requirements for LWR Nuclear Power Plants (2001). The joint project was planned to allow a first assessment of this concept in September 2009, which could be achieved through synergies within the Generation IV International Forum. The project plan has been outlined by Starflinger et al. (2007). It is structured into working packages for design integration, core design, safety systems, materials, heat transfer, as well as education and dissemination, to enable mapping with the project structure of the Generation IV International Forum, i.e. system integration and design, thermal-hydraulics and safety, materials and chemistry. Meanwhile, the number of European partners contributing to this project has been increased by the universities of Delft and Budapest and by the Joint Research Centre in Petten as active supporters. Moreover, the universities of Pisa, Manchester and Lappeenranta, and may be even more in the near future, are contributing outside the project, since the importance of this project for nuclear education and training became obvious.

On a closer look, the core of the HPLWR has to solve a lot more design challenges than simply an increase of the core exit temperature by around 200 °C. Assuming a typical feedwater temperature of 280 °C like with supercritical fossil fired power plants, the enthalpy rise in the core would exceed the one of a conventional pressurized water reactor with 35 °C coolant heat up almost by a factor of ten. A conventional core design with a single stage coolant heat up from bottom to top would result in peak cladding temperatures beyond any reasonable cladding material limits, if all power and mass flow non-uniformities, uncertainties and tolerances as well as allowances for operation are taken into account. Ideas to solve this issue can be found at coal fired boilers. There, the coolant is typically heated up in three steps, namely the evaporator (which means the transition from liquid like to steam like conditions at supercritical pressure) and the first and second superheaters with higher temperature but lower power when approaching the envisaged boiler outlet temperature. Intensive coolant mixing between each step eliminates hot streaks of the preceding step before entering the next one. As an example, Schlenberg et al. (2008) proposed a thermal core concept in which the evaporator assemblies are placed in the centre of the core, followed by the first superheater assemblies with downward flow surrounding them, and the second superheater assemblies with upward flow at the core periphery, where the fissile power is low anyway because of neutron leakage. The European consortium decided in September 2006 to take this example as a basis for their joint core design study, but to work in parallel also on a fast core option with limited manpower.

Besides the core design and analysis, the project includes the design concepts of the reactor pressure vessel, the containment with its safety systems, as well as major components of the balance of plant including first analyses of them to assess realistically costs and safety features of the power plant at the end of the project. The project is accompanied by cladding material tests and detailed heat transfer studies which were identified as key technologies of supercritical water cooled reactors by the Generation IV International Forum.

The first year of the project, which lasted until September 2007, concentrated on the mechanical design of core and reactor components, prepared the required codes for this new application and defined material properties to provide tools and detailed information for several analyses which were performed then in the second year. Starflinger et al. (2008) summarize results of the first year. Results of the second year were reviewed in a mid-term assessment in September 2008. There, the project partners got the first impression about the feasibility of the new core concept and provided suggestions for further improvements. Starflinger et al. (2009) report about further details. The following chapters shall illustrate some more details of the working packages and document the scientific and technological advances.

2. Core design

2.1. Thermal core design

Most work has been performed for a core with a thermal neutron spectrum. It is based originally on an assembly cluster design by Hofmeister et al. (2007) consisting of 9 smaller assemblies with 40 fuel rods each, surrounded by an assembly box and with an additional moderator box in the centre. The fuel rods have an outer diameter of 8 mm and a pitch to diameter ratio of 1.18. As a new feature, wire wraps are used as spacers providing good coolant mixture in upward and downward flow directions. Additional moderator water is provided in the 9 mm gaps between the assembly boxes as well as inside the moderator box. Herbell et al. (2008) propose to use a honeycomb structure with thermal insulation to minimize the box material and to avoid pseudo boiling of the moderator water. Fischer et al. (2007b) describe the common head and foot pieces of each cluster as well as the mixing chambers above and underneath the core. 52 of these assembly clusters are dedicated to the evaporator, the first and the second superheaters each. Control rods are assumed to run inside the moderator boxes. Their reactivity feedback has been discussed by Schlagenhauf et al. (2007). The core power distribution has been analyzed by Maráczky et al. (2008) using the two group diffusion code KARATE which was coupled with the thermal-hydraulic code SPROD. Results shown in Fig. 1 indicate the intended highest power density in the evaporator and the lowest power in the second superheater, achieving an average power density of around 60 MW/m³. There is certainly still more room for optimization which implies, however, more work than needed for a conventional core design. Similar first results were achieved by Monti et al. (2008) with the neutron transport code ERANOS coupled with the system code TRACE. The wire wrap is providing sufficient coolant mixing inside assemblies as demonstrated by Himmel et al. (2008) with an innovative sub-channel analysis shown in Fig. 2. It has been confirmed by CFD analysis of a short segment of a fuel assembly, performed by Kiss et al. (2009). Coolant mixing in the upper mixing chamber has been predicted by Wank et al. (2008), resulting in a need for additional mixing devices to eliminate hot streaks emitted from individual evaporator assembly clusters. As density wave oscillations might occur in the evaporator assemblies, like in a boiling water reactor, Ortega Gomez et al. (2008) propose to use inlet orifices for each assembly to keep sufficient margin from stability limits.

Based on these and further preliminary results, the final project review in September 2009 concluded that the thermal core concept should be feasible in principle, though being highly innovative. Suggestions for further improvements like a change of the moderator
flow structure, a reflector design with an additional water layer, mixing devices in upper and lower mixing chambers, and an even better thermal insulation of moderator and assembly boxes were taken into account during the project. Moreover, burn-up analyses and detailed hot channel analyses confirmed that the target peak cladding temperature of 630 °C can be met even under worst case assumptions.

2.2. Fast core option

Basically, the core design could be simplified significantly by omitting all moderator water channels such that the neutron spectrum becomes fast or at least epithermal. Moreover, a conversion ratio close to one or even a breeding ratio greater than one would result in a more efficient use of fissile material. Core design studies by Mori (2005) which were confirmed and extended later by Rimpault (CEA, unpublished) revealed, however, that a negative coolant void coefficient is not easy to assure throughout the entire burn up cycle. Only a very heterogeneous arrangement of seed and blanket assemblies, together with additional solid moderator layers and the use of multiple heat-up steps could help to achieve a core configuration with safe reactivity feedback mechanisms. As such design optimization would exceed the planned scope of the project by far and, on the other hand, significant work is performed currently by the University of Tokyo, as summarized by Ishiwatari et al. (2008), the consortium decided instead to rely on the synergies in the Generation IV International Forum and to concentrate rather on the thermal option for the rest of the project.

3. Reactor pressure vessel

The reactor pressure vessel design concept (Fig. 3) is based on the latest design of pressurized water reactors with control rods supplied from the core top, but with a stronger wall thickness of 45 cm in the cylindrical part to account for the higher system pressure, as well as a thermal insulation of inlet and outlet flanges to minimize thermal stresses. The 4 feedwater lines enter the pressure vessel through backflow limiters (Fig. 4) to avoid a temporary flow reversal in the core in case of a large break of one of the feedwater lines. Their flow characteristic has been analyzed by Fischer et al. (2007a) showing that the pressure loss in reverse flow direction is more than 10 times higher than in normal direction. Thermal
Fig. 2. Coolant temperature profile in the hottest superheater assembly with 20% assumed radial power gradient [Himmel et al., 2008].

Fig. 3. Reactor pressure vessel design with 3 assembly clusters in the core representing (from right to left) an evaporator, a first and a second superheater cluster.

Fig. 4. Design of the backflow limiter at the feedwater inlet [Fischer et al., 2007a].

Stresses and deformations of the pressure vessel and of the steam plenum have been checked by Fischer et al. (2008). As the thick walled reactor pressure vessel is only exposed to colder feedwater at around 280 °C, and all components exposed to superheated steam have been designed with sufficient distance from the thick walled structure, as shown in Fig. 5, the stresses and deformations are indeed sufficiently low despite the higher pressures and temperatures of this reactor concept. Accordingly, the project review in September 2008 confirmed that the pressure vessel design is expected to be acceptable, but requested for further transient analyses to check fatigue and crack initiation.

4. Balance of plant concept

Like in a supercritical fossil fired power plant, the superheated steam with 500 °C shall be supplied directly to the high pressure
(HP) steam turbine. **Fig. 6** shows the actual steam cycle concept for the balance of the plant. Part of the steam is extracted to reheat the steam after expansion in the HP turbine at 4.25 MPa to 441 °C temperature, without the need of steam separators, to be supplied to the intermediate pressure (IP) and low pressure (LP) turbines afterwards. River water at 15 °C is assumed to condense the steam at 5 kPa and 33 °C. Condensate pumps and preheaters (PH5–PH7 in **Fig. 5**), using steam extractions from the LP turbine, heat the feedwater up to 135 °C. A feedwater tank at 0.55 MPa also serves as a preheater using steam extracted from the IP turbine exit. From there, 4 high pressure feedwater pumps, of which 3 are needed for the full mass flow rate of 1179 kg/s and a number 4 pump which is kept on hot stand by, produce a pressure of 25 MPa at the reactor inlet. Another 4 preheaters are needed finally to reach the envisaged feedwater temperature of 280 °C. **Brandauer et al.** (2009) predict a net efficiency of this steam cycle of 43.5%.

In the envisaged load range from 500 MW to 1000 MW net electric power, the turbine control valve shall keep a constant reactor inlet pressure of the feedwater at 25 MPa. Feedwater mass flow is controlled with the feedwater pumps, while the temperature of the superheated steam is kept constant at 500 °C by the control rods of the core. A control valve for the extracted steam is keeping the temperature of the reheated steam constant at 441 °C. The plant control has been modelled with the system code APROS by **Schlagenhauger et al.** (2009). A first turbine design concept worked out by **Herbell et al.** (2009) assumes that a full speed turbine rotor (50 Hz) can be used instead of the half speed rotor design of saturated steam turbines needed for conventional light water reactors, indicating already significant cost savings for the balance of plant.

The start up concept of the steam cycle includes separators and control of the supercritical pressure with a valve instead of the turbines.

### 5. Safety systems

Even though the HPLWR plant concept looks similar to a boiling water reactor (BWR) facility, at a first glance, it differs fundamentally by the missing primary pumps. Whereas a control of water inventory in the reactor pressure vessel is sufficient for the BWR to ensure the residual heat removal even in case of accidents, a continuous coolant mass flow rate through the reactor is required for this once through steam cycle as there is no closed coolant loop inside...
the reactor. This can be achieved either with redundant feedwater pumps or by depressurization of the pressure vessel such that the residual heat is removed by vaporization. In case of accidents, these functions must also be provided inside the containment. With this respect, most safety systems of the containment can indeed be derived from latest BWR containment concepts, with the exception of passive flooding and emergency condensers for reasons explained above.

A first design proposal for such a containment has been sketched by de Marsac et al. (2008), shown in Fig. 7. Containment isolation valves for each of the 4 feedwater and steam lines, inside and outside of the containment, close automatically in case of a feedwater or steam line break inside or outside the containment. The reactor is scammed and the depressurization valves release steam through 8 spargers into 4 upper pools, removing the residual heat until at least one of the 4 redundant, active low pressure coolant injection pumps in the basement of the containment will start. In case of a steam line break inside the containment, any pressure increase by steam release is limited by a large pressure suppression pool in the lower half of the containment which is connected by 16 open pressure suppression tubes. As an additional passive safety system, de Marsac et al. (2008) propose to use steam injectors to supply feedwater with high pressure from coolers, hanging in the upper pools and driven by steam produced in the core during depressurization. An overflow line from the spargers is starting the steam injectors within the first 10 s. This design proposal, however, still needs to be verified. As a back-up alternative to cool down the core without steam release to the containment, emergency condensers in the upper pool could be connected with the steam and feedwater lines, supplying the condensate to the core through a motor driven recirculation pump. Long term passive residual heat removal from the containment shall be provided by containment condensers to the spent fuel pool above the containment.

6. Cladding material and water chemistry tests

Corrosion tests at supercritical pressure and temperatures up to 650 °C are being performed at VTT, supported by autoclave tests at JRC Petten and KFKI-Atomic Energy Research Institute. Candidate cladding alloys under examination are ferritic–martensitic steels, stainless steels, nickel-base alloys and ODS alloys. Recent results have been published by Penttilä et al. (2008). As expected, the corrosion resistance increases with increasing chromium concentration, as summarized in Fig. 8, with a compromise however to the strength of these alloys. Therefore, stress corrosion crack tests and tests of the tensile strength are performed in autoclaves at supercritical pressure in parallel. Creep rupture tests followed in 2009. Up to now, however, an appropriate cladding alloy could not yet be identified which meets the design target.

In-pile tests of radiolysis and its impact on the water chemistry and corrosion are being prepared at CVR Research Center in Rež. A supercritical water loop with an active channel inside the LVR-15 reactor has been completed. It is currently tested out of pile to be commissioned for in pile operation after 2010. The auxiliary unit with heaters and coolers, the purification system, water chemistry monitoring and sampling, and the dosing system are shown in Fig. 9. Details of this system have been described by Hájek et al. (2007).

The partners share their test results with research institutes and industrial partners in Japan, Canada, Korea, and France. The project agreement on materials and chemistry, which has been worked out jointly within the Generation IV International Forum to be signed in 2010, includes in-pile tests and a base material development program for corrosion and creep resistant claddings in Japan, coating tests in Canada to protect a creep resistant base metal at elevated temperatures, and water chemistry tests in Canada.

7. Heat transfer studies

In general, supercritical water undergoes strong changes in its thermo-physical properties near the pseudo-critical point at a given pressure. Such changes strongly influence the heat transfer phenomena in such a fluid resulting in normal, deteriorated or enhanced heat transfer rates, see e.g. Yamagata et al. (1972) and Koshizuka et al. (1995). At a high heat flux and a low mass flux, the heat transfer can deteriorate causing local hot spots which cannot be predicted with any correlation at all up to now. Consequently, improved or even new heat transfer correlations are necessary for the design of HPLWR core. The HPLWR project
partners are trying to suitably address this issue with detailed Reynolds Averaged Navier Stokes (RANS) based CFD analyses. Palko and Anglart (2008) succeeded to model the deterioration of heat transfer of supercritical water in vertical tubes by using the Shear Stress Transport (SST) turbulence model and a very fine resolution of the boundary layer, with the dimensionless distance of the mesh closest to the wall $y^+ < 1$. Their convincing simulation of an experiment of Shitsman (1963) demonstrates that buoyancy forces were responsible for local hot spots and need to be included in proper modelling, see Fig. 10.

Zhu and Laurien (2008) used data of the SPHINX facility at KAERI (Kim et al., 2006) to validate their numerical studies of heat transfer in supercritical CO$_2$, which is another example of collaboration in the Generation IV International Forum. Results of Visser et al. (2008) confirm as well with these data that heat transfer phenomena in tubes can be predicted quite satisfactorily if the boundary layer is resolved with $y^+ < 1$.

The influence of the presence of a wire-wrap on the heat transfer with supercritical fluids is first analyzed in simple geometries using the adopted RANS based CFD approach; see e.g. Chandra et al. (2009). This reveals that the consequences of heat transfer deterioration are mitigated by the presence of a wire wrap. As the next step, the tested methodology has been applied to typical HPLWR fuel assembly geometries with wire wraps. As an example, the influence of the presence of a wire-wrap in a four rod-bundle has been studied. The surface temperature of a four rod bundle, predicted with CFD for hot-channel conditions simulating an evaporator assembly with supercritical water, is shown in Fig. 11. The heat flux was assumed to be 1375 kW/m$^2$ and the mass flux 1332 kg/m$^2$ s. The presence of a wire-wrap around the fuel rods enhances mixing...
in this four rod bundle geometry and helps to make the temperature distribution of the cladding more uniform in comparison with bare fuel rods. The CFD results show that the average heat transfer rate from the fuel rod surface to the supercritical fluid is typically increased by 10% by means of the wire wrap. First results also indicate that the onset of heat transfer deterioration could be prevented. However, this result still needs more studies for further confirmation.

As heat transfer tests with supercritical water could not yet be afforded within the scope of this project, several component tests with supercritical Freon in Japan, with supercritical CO₂ in Korea and Canada, and finally supercritical water tests of small scale bundles in Japan and China shall validate the numerical predictions of the European HPLWR project. An associated project agreement on thermal-hydraulics and safety with Canada and Japan has been signed in October 2009.

8. Education and training

Within three years, the European project on the HPLWR has reached an enormous progress. This is not only due to the excellent financial support of the European Commission but also due to the enthusiasm of the young generation of nuclear engineers who selected this project as a training field to learn about innovative light water reactor technologies. Since the beginning of this project, more than 20 bachelor or master theses and more than 10 doctoral theses have been completed at partners of this project, documenting that nuclear engineering is still one of the most fascinating subjects of power plant technologies. Almost all of these students were staying in the nuclear business afterwards where they are filling at least partially the urgent needs for new recruitment at the nuclear industry and research.

An international students’ workshop on High Performance Light Water Reactors, which was held in April 2008 in Karlsruhe, gathered 30 students and young scientists from European partner organizations, from GIF members in Japan and Canada as well as from interested research organizations in China and India. As summarized by Anglart et al. (2008), they presented a status of their master or doctoral thesis in relation with supercritical water cooled reactors in a 4 day seminar, accompanied by lectures of professors of the HPLWR project who provided an overview and background information.

9. Dissemination of results

Being in a pre-competitive stage of the development of this reactor concept, the partners of this project are actively publishing all new results at conferences and in journal papers to stimulate discussions, to ask for critical feedback and to derive jointly new ideas to meet the challenging targets. In total, more than 100 papers were produced and presented in the 3 years since the beginning of the project. An ideal platform for discussions is the International Symposium on Supercritical Water Cooled Reactors which is held every second year in one of the member states of the Generation IV International Forum. In March 2007, the Shanghai Jiao Tong University took the opportunity to invite all partners working on SCWR subjects and thus to introduce China as a new GIF member to the forum. In total, 59 papers were presented about design and technologies of SCWR. Having plenty of results available after 2 years of intensive studies, the HPLWR consortium invited in turn all partners of the forum, but also industry, research institutes and students who had not yet been involved in this technology, to the 4th Symposium on SCWR design and technologies held in Heidelberg in March 2009. There, the number of papers and presentations had increased already to more than 80, while more than 100 participants were listening to the numerous presentations.

10. Outlook

The overall, ambitious target of this project is to complete the conceptual design of the reactor, the containment, the safety systems and the balance of plant before the end of 2009 such that it can be assessed with respect to the Generation IV criteria, saying that the power plant shall be more economic, reliable, and at least as safe and as proliferation resistant as the 3rd generation of light water reactors, while producing less nuclear waste. This implies that the functionality of safety systems must be studied with transient analyses, start-up and shut down procedures must be developed, and even the first layout of the steam cycle components must be sketched to produce a reliable data base for the assessment.

Despite all fascinating results, however, we must not forget that there is still a long way to go before a HPLWR could be built. We have to be aware that the design concepts will still be iterated several times before a convincing status will be reached. Moreover, component tests will be needed to confirm what design and analyses predicted and initial concepts might be withdrawn or modified afterwards. The next milestone beyond the scope of this project is considered to be the in-pile test of a single assembly cooled with supercritical water at realistic pressures and temperatures. This test will not only validate the computer codes used for reactor design and force the scientists to documented quality assurance of their predictions, but also challenge the regulators to license the first nuclear system operated with supercritical water. In this context, the Generation IV International Forum with its projects on Supercritical Water Cooled Reactors is considered as an ideal platform to exchange information and ideas, to share test results, to benchmark codes and, finally, to collaborate on component tests or even a prototype test reactor.

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