DESIGN CONCEPT OF A SUPER PRESSURIZED WATER REACTOR

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Abstract

The Super Pressurized Water Reactor (SPWR) is meant as a near term application of Supercritical Water Reactor Technologies to Pressurized Water Reactors (PWR) of the third generation. It shall be operated at a supercritical pressure of 25 MPa, but with a core outlet temperature of 370°C, which is colder than the pseudocritical temperature such that the coolant will still be pseudo-liquid there. This will require again a closed primary system with steam generators, a pressurizer and reactor coolant pumps. The higher core outlet temperature, however, would enable an increase of the net electric power by more than 100 MW and an increase of the net efficiency by around 9 %, over-compensating by far the higher steel costs due to thicker walls. Trying to minimize the technical risks of such evolutionary reactor development, the concept proposed here is based on the proven design of reactor internals of the EPR. The maximum cladding surface temperature of 410°C, however, will be challenging for advanced Zircalloy materials and stainless steel claddings might be required instead. Moreover, the lower coolant density will require some core design modifications. Aiming at 1700 MW net electric power, four once-through steam generators, instead of conventional U-tube steam generators, can produce slightly superheated steam in the secondary system, without exceeding a tube sheet thickness of 650 mm despite the higher pressure.

Keywords: Power Plant Design Concept, Core Design Concept, Steam Cycle Analysis, Near Term Application

1. Introduction

Since the 1980s, when Pressurized Water Reactors (PWR) with more the 1300 MW_e were built in series and we considered them already as mature products, they still got several further improvements during a continuous development process. The fuel rod diameter was stepwise decreased to 9.5 mm today, and the number of fuel rods in the core was increased, which increased the surface heat flux as well as the core power. The enrichment was increased to 5 % today and gadolinia was added to the fuel to compensate the excess reactivity of fresh fuel, resulting in a burn-up of more than 60 MWd/t_{HM}. Advanced turbine aerofoils could enhance the turbine efficiency. Since 2000, the safety systems were prepared for severe accident mitigation such that an evacuation of the population around the power plant would not be needed anymore even in case of a core melt-down. Finally, more passive safety systems were introduced to gain more grace time for the operators in case of severe accidents.

Compared with these improvements, the increase of temperature and pressure of the steam cycle stayed very limited, which had been the key driver for power and efficiency improvements in fossil fired power plants. Innovative concepts of SuperCritical Water-cooled Reactors (SCWR), which were studied e.g. in Japan, Europe, and Canada [1], resulted in live steam temperatures of 500°C or more, but the core design had to be quite different. Multiple heat-up steps with intermediate coolant mixing were foreseen to avoid unrealistic material temperatures of fuel claddings, quite different fuel shuffling schemes had to be introduced, and new materials had to be developed. An entirely new core design, however, can hardly be tested prior to installation, and these concepts are rather to be considered as visionary goals today. They are not expected to be built in near future.

With this background, we like to propose an evolutionary design concept for PWRs, which is not as ambitious as the SCWR, but which is based on proven PWR technologies as far as possible. It shall allow an increase of live steam temperature by more than 50°C, resulting in a higher steam cycle efficiency and, accordingly, in more electric power from the same core power. Let us call it the Super Pressurized Water Reactor (SPWR).

2. Core Design Concept

A higher live steam temperature can only be achieved with a higher core outlet temperature which, in turn, requires a higher operating pressure of the primary system. Let us assume that we stay with the same core design and the same reactor internals like a third generation PWR, e.g. the EPR [2]. It has 241 fuel assemblies with an array of 17×17 rods of 9.5 mm outer diameter, of which 89 assemblies are equipped with 24 control rod fingers each. The 265 fuel rods per assembly have an active length of 4.2 m. The total length of the fuel assembly is 4.8 m.

2.1 Thermal hydraulic concept

The operating pressure shall be increased, however, from 15.5 MPa to 25 MPa. This pressure is higher than the critical pressure of water and the surface tension is zero. Around a temperature of 384°C, known as the pseudo-critical temperature at 25 MPa, the fluid properties change continuously from liquid to steam without forming bubbles or droplets. If we choose a core inlet temperature of 345°C and a core outlet temperature of 370°C, the coolant will still be liquid-like at the core outlet. The coolant enthalpy will increase then by about 200 kJ/kg, in average, like in the core of the EPR. Again, the core power distribution cannot be uniform, but neutron leakages, control rods, fuel shuffling, neutron poisoning and burn-up effects are causing hotter and colder fuel rods. Moreover, the coolant mass flux cannot be uniform either. Let us assume, therefore, that the coolant mass flow through the hottest subchannel of the reactor core is receiving twice this enthalpy increase. This will cause a peak coolant temperature of 381°C, which is still liquid-like, even rather close to the average coolant outlet temperature, as the specific heat of water at 25 MPa has a pronounced peak at the pseudo-critical temperature, keeping the temperature increase limited.

The maximum cladding surface temperature to be expected can be discussed with Fig. 1. Like a Nukiyama diagram, it shows the heat flux versus the surface temperature superheat of the claddings, compared with the bulk temperature, exemplarily for a mass flux of 3000 kg/m^{2}s and a hydraulic diameter of 12 mm. For two sub-critical pressures of 15 and 20 MPa, we assumed the bulk fluid to be saturated liquid. For the two supercritical cases, we assumed a pressure of 25 MPa and a bulk temperature of 360°C, representing an average sub-channel, or 380°C, representing a hot sub-channel. For subcritical conditions, the curves are constructed from the Dittus-Boelter [3] correlation for subcooled fluids, the Rohsenow correlation [4] for the nucleate boiling regime, the Groeneveld et al. [5] look-up table for the critical heat flux, and the film-boiling look-up table [6] for the film boiling regime. For supercritical conditions, we took the trans-critical look-up table of Zahlan [7]. With a maximum linear heat rate of the fuel rods of 41 kW/m, which should not be exceeded to avoid damage in the fuel centreline, and a fuel rod diameter of 9.5 mm, we get a maximum heat flux of 1374 kW/m^2, which is indicated with a dotted line in this diagram. At 15 MPa, this is not far anymore from the critical heat flux of 2140 kW/m^2, from which we have to keep a respectful margin to avoid Departure from Nucleate Boiling (DNB) and burn-out. At 20 MPa, this margin would become zero, which explains why just a small increase of pressure is not advisable. At supercritical pressure, however, the DNB phenomenon disappears. Instead, we get a maximum at a much higher heat flux of about 3600 kW/m^2, which is known as the onset of deteriorated heat transfer. This will allow smaller fuel rod diameters, higher heat flux and higher power densities than in a conventional PWR. As a drawback, however, the wall temperature superheat will be higher. For the hot subchannel, we read from the diagram a wall temperature superheat of 30°C, resulting in a maximum cladding surface temperature of 410°C. This will certainly not be a problem for stainless steel claddings, which are under discussion for the SCWR, but it will be a challenge for Zircalloy claddings, which are still preferred for their excellent neutron transparency.

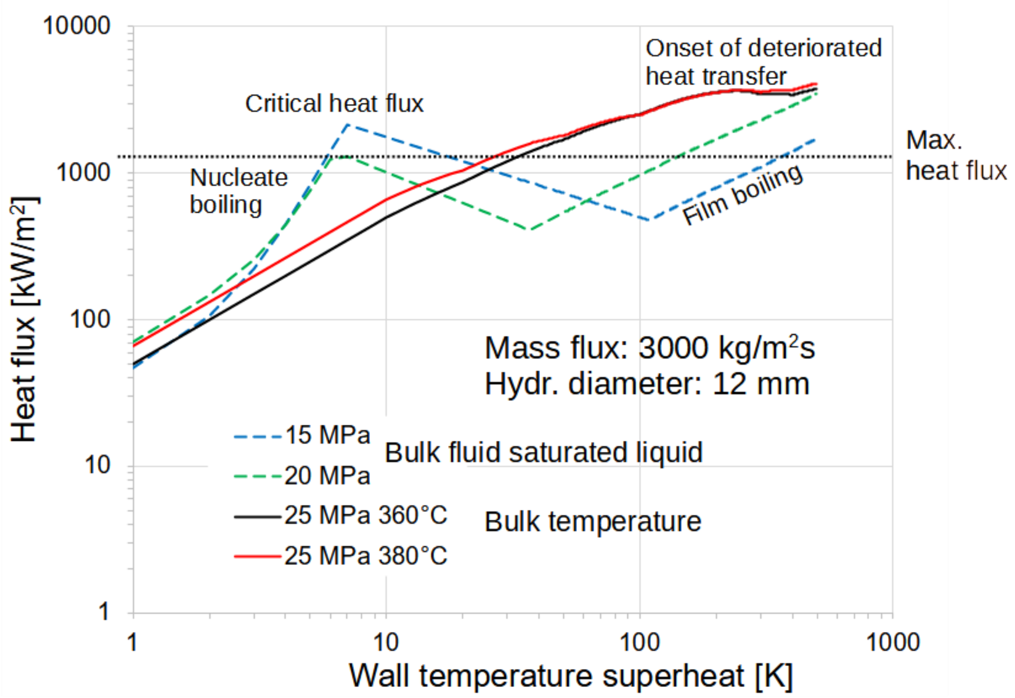

Fig. 1: Predicted heat transfer at fuel claddings

2.2 Neutronic concept

In some neutronic aspects, the SPWR would behave rather similar to the EPR. As the dimensions of the core and fuel assembly lattice would not change significantly, the reactor physics behaviour would be similar in many aspects. We believe that the reactivity control system, well proven for EPR, could be used without need of significant redesign. Burnable absorbers could also be used to partially compensate for the large reserve reactivity at the beginning of a burnup cycle, and, along with enrichment profiling, for making the power density profiles within a fuel assembly as flat as possible.

Nevertheless, there are some challenges that will have to be considered. The most important ones are listed below.

- With an average coolant temperature of 311.5°C at a pressure of 15.5 MPa of the EPR, the average moderator density is approximately 0.702 g/cm^3. On the other hand, due to the higher average coolant temperature of 360°C and higher pressure of 25 MPa, this density is 0.591 g/cm^3 for the SPWR. Accordingly, the fuel lattice parameters might need some modification, shifting the optimal moderator-to-fuel volume ratio to larger values.
- According to the significantly lower moderator density, the safety relevant Moderator Temperature Coefficient (MTC) will have to be examined carefully so that under-moderation and/or negative MTC can be assured for all parameter values along a burnup cycle.

- We expect that boron control will be possible in the SPWR, similarly to the EPR. Due to the lower moderator density, however, the reactivity effectiveness of the dissolved boron will be somewhat decreased. This could be compensated by increased boric acid concentrations, paying close attention to not making the MTC positive.
- Probably the most significant influence in a neutronic point of view, however, would be the use of stainless steel cladded fuel instead of Zircalloy. Stainless steel has rather unfavorable neutron absorption properties, i.e. the thermal absorption cross section is substantially higher than that of zirconium. Nevertheless, if zirconium alloys are not possible to use due to higher than usual PWR cladding temperatures, the significant loss of reserve reactivity would have to be compensated by use of higher enrichments. This would, most probably, induce the application of higher than 5 % enriched UO_2.
- Due to the lower moderator density, the diffusion length of thermal neutrons will be somewhat different. Accordingly, the behavior of the reactor with respect to spatial xenon oscillations may be slightly different from that of EPR.
- Due to the expected harder neutron spectrum, it is also expected that the evolution of transuranic nuclides, such as plutonium isotopes, may be different from that of EPR to a small extent. This, of course, depends significantly on whether zircalloy of stainless steel will be used as cladding and on what modifications are made to lattice parameters of the fuel.

3. Primary System

Staying with the EPR core design and its reactor internals, let us keep the four coolant loops as well. A reactor pressure vessel for this supercritical pressure has already been designed by Fernandez et al. [8]. It has a wall thickness of 370 mm in the cylindrical part, with reinforcements up to 472 mm around the inlet and outlet nozzles.

The pressurizer can be designed as for subcritical pressures, except the stronger walls, which are needed due to the higher pressure. At supercritical pressure, the temperature in the pressurizer is kept at pseudocritical temperature, such that a small change of temperature will cause a large change of density, which in turn changes the pressure in the primary system. Different from subcritical pressure, however, the liquid level in the pressurizer cannot be controlled anymore by the mass flow from the makeup water system. Instead, the electric heaters or spray coolers inside the pressurizer are controlled by the temperature and the mass flow of supplied water is controlling the pressure.

No significant changes are expected for the reactor coolant pumps either, except that they need a stronger pump casing and the shaft sealing system must be designed for a higher pressure.

The four steam generators, however, must be different. The limiting part of the steam generators are the tube sheets, which need to carry a higher design pressure now. Inside conventional U-tube steam generators, however, the thickness of these tube sheets is already at its limit of 650 mm. To increase the pressure of the primary system from 15.5 to 25 MPa, the total cross section of the tube bundle must be reduced to 15.5/25 = 62 % to keep the same thickness. This can be achieved with once-through steam generators with straight tubes, as sketched in Fig. 2, instead of U-tube steam generators. The coolant of the primary system enters from top, passes through an upper tube sheet (indicated in red) into the tube bundle, and leaves through a lower tube sheet at the bottom. Each of these tube sheets is carrying now half the number of tubes and the thickness can be kept again below 650 mm. An additional advantage is that steam separators and dryers can be omitted and the steam can even be slightly superheated. An advantage of conventional U-tube steam generators, on the other hand, was that tubes can freely expand without thermal stresses.

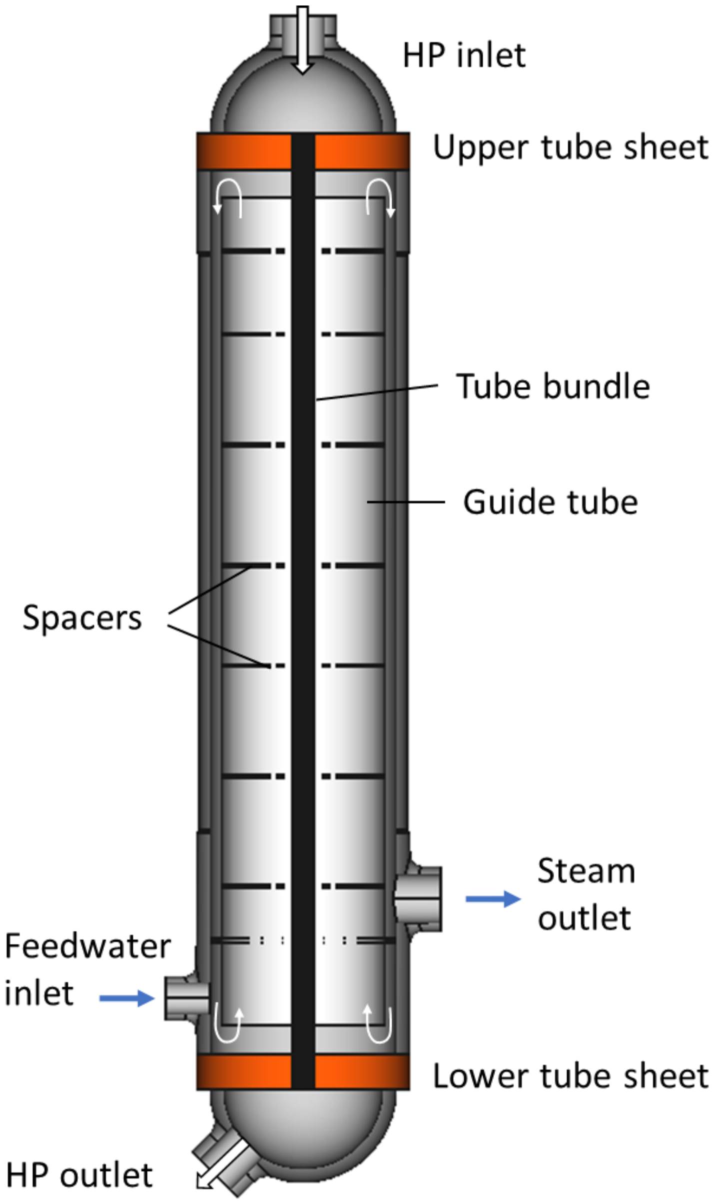

Fig. 2: Sketch of a once-through steam generator with straight tubes.

Such once-through steam generators are less common, but not entirely new. Two of them, used instead of four U-tube steam generators, were installed already in 1986 in the power plant Mülheim-Kärlich, Germany, built by Babcock-Brown Bovery Reaktor GmbH (BBR), a PWR with 1300 MW_e [9]. Unfortunately, this power plant was operated only for three years, so that we cannot report about long term experience today. Such once-through steam generators, however, are still manufactured by BWXT Canada Ltd. and offered for replacement to nuclear power plants [10].

A drawing and some design data of the BBR once-through steam generators can be found in [9]. The steam generators had a total height of 23 m, each with a tube bundle of 16000 tubes made of Inconel 600, with 16 mm outer diameter and 0.9 mm wall thickness, and with a length of 16 m. The hexagonal tube array had a pitch to diameter ratio of about 1.35. The tubes were heated with a mass flow of 9500 kg/s of primary coolant of 15.5 MPa, with an inlet temperature of 329°C and an outlet temperature of 296°C. On the steam side, feedwater with a mass flow of 1000 kg/s was entering with 236 °C and the steam was superheated by 27 °C to 312°C at 6.9 MPa. The superheated steam was guided by an internal tube around the bundle along the inside of the pressure vessel (made of 22NiMoCr3-7) to keep it warm. As the tubes were welded into the tube sheet on both sides, they cannot expand freely when they are heated. To avoid tube bending, the shell temperature must be kept hot enough such that tubes and vessel get a similar thermal expansion during operation or at least cause tensile stresses inside the tubes. The steam outlet at 2.8 m above the lower tube sheet, as shown in Fig. 2, was keeping the upper shell at superheated steam temperature to achieve this.. In addition, several spacers were needed at non-uniform distances to avoid tube vibrations and to keep the tubes straight.

Using the lookup tables and heat transfer correlations as in Fig. 1, the temperature distribution inside this steam generator can be predicted as shown in Fig. 3. The lowest 2 m of the tube bundle form the economizer, in which the feedwater is preheated to saturation temperature. As long as the tube wall is wetted, from 2 to 8 m height, the heat transfer on the steam side is excellent, and the heat resistance in mainly due thermal conduction in the tube wall. At a distance of 8 m, measured from the lower tube sheet, 80% of the feedwater has already been evaporated. Here we see a dryout, with a sudden decrease of heat flux and a step increase of the wall temperature. From there on, the heat resistance is mainly due to the poor heat transfer on the steam side. Consequently, the entire upper half of the steam generator is needed to fully evaporate and superheat the steam. The small step in heat transfer at saturated steam conditions (at 11.5 m) is just due to a mismatch of the film boiling lookup table with the Dittus-Boelter correlation for superheated steam. It does not have any physical meaning. As a result, the steam can indeed be superheated to 317°C, leaving some margin for the downward path to arrive with 312°C at the steam outlet.

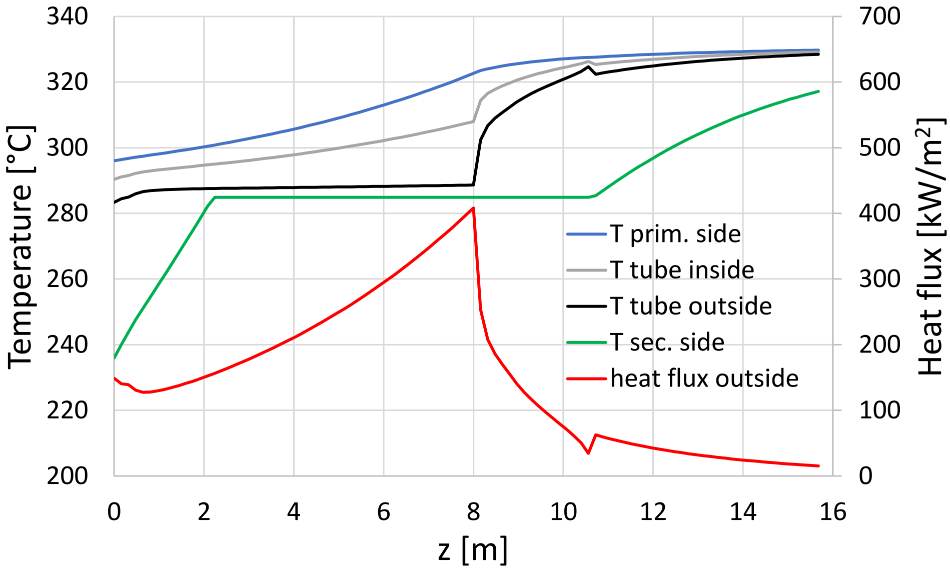

Fig. 3: Simulated temperatures of the once-through steam generator of BBR.

4. Secondary System

A suitable steam cycle for an SPWR can be dimensioned again based on those of conventional PWRs. As an example, we took here the secondary system of the AP1000 [11], which was simplified a bit to serve as a reference steam cycle, as shown in Fig. 4. Saturated steam at 5.55 MPa is expanded first in a High Pressure (HP) turbine and is reheated then in a Moisture Separator and Reheater (MSR) to 256°C at 1.10 MPa, before it is expanded in three Low Pressure (LP) turbines to the condenser pressure. The condensate is pressurized first in Condensate Extractions Pumps (CEP), followed by four LP preheaters, which are condensing extracted steam. The deaerator is kept at 1.09 MPa by steam extracted from the outlet of the HP turbine. Two-stage feedwater pumps are increasing the feedwater pressure further to 8.65 MPa, to be preheated by two HP steam extractions to 228°C. Taking all pressures and temperatures from the AP1000, the mass balances of all components form a linear equation system, which was solved with MATLAB. Liquid and steam properties were determined with XSteam [12]. The result is shown in a temperature-entropy-diagram (T-s-diagram), Fig. 5. The net efficiency of this steam cycle, defined here as the turbine power minus the power of condensate extraction pumps, feedwater pumps and reactor coolant pumps, divided by the core power, is 35.6%.

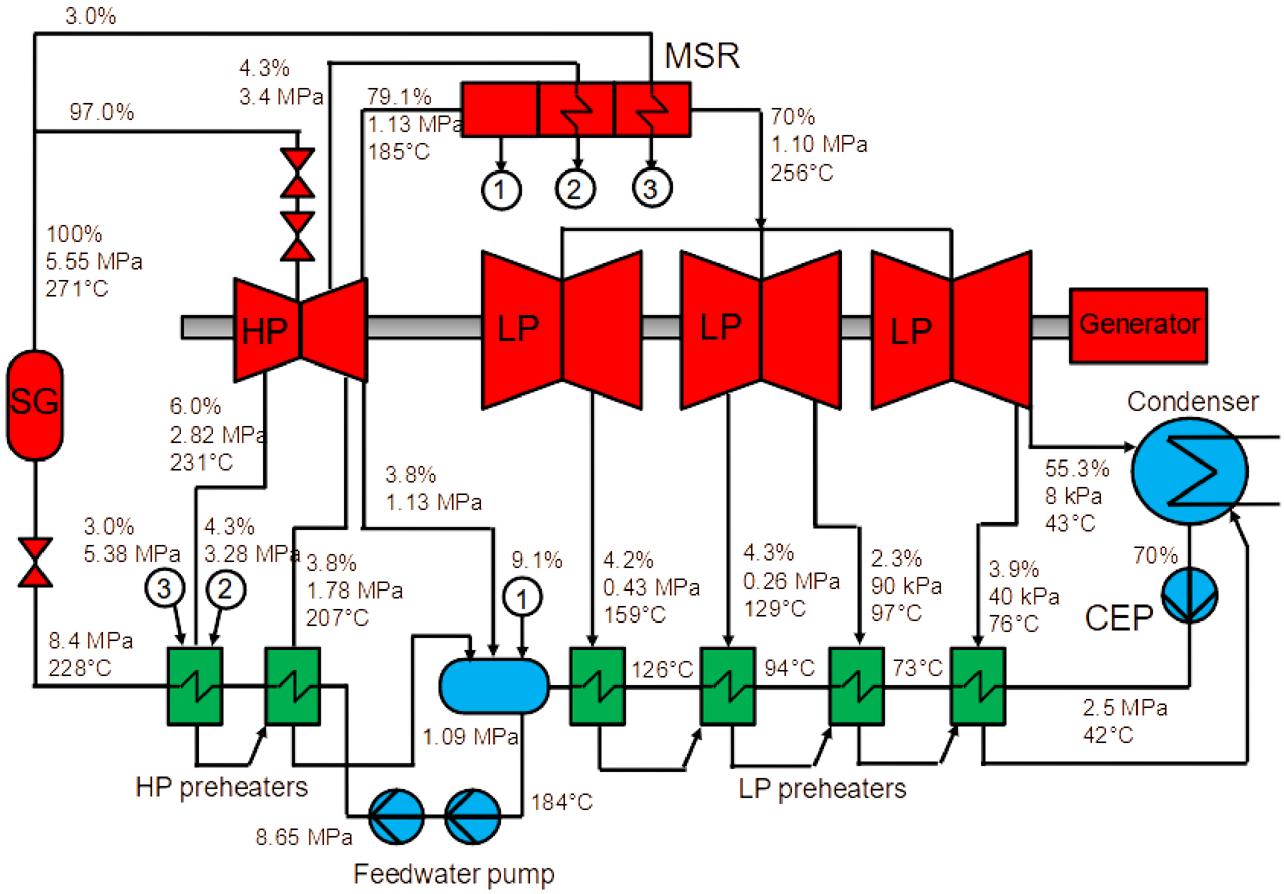

Fig. 4: Secondary system of a conventional PWR as reference.

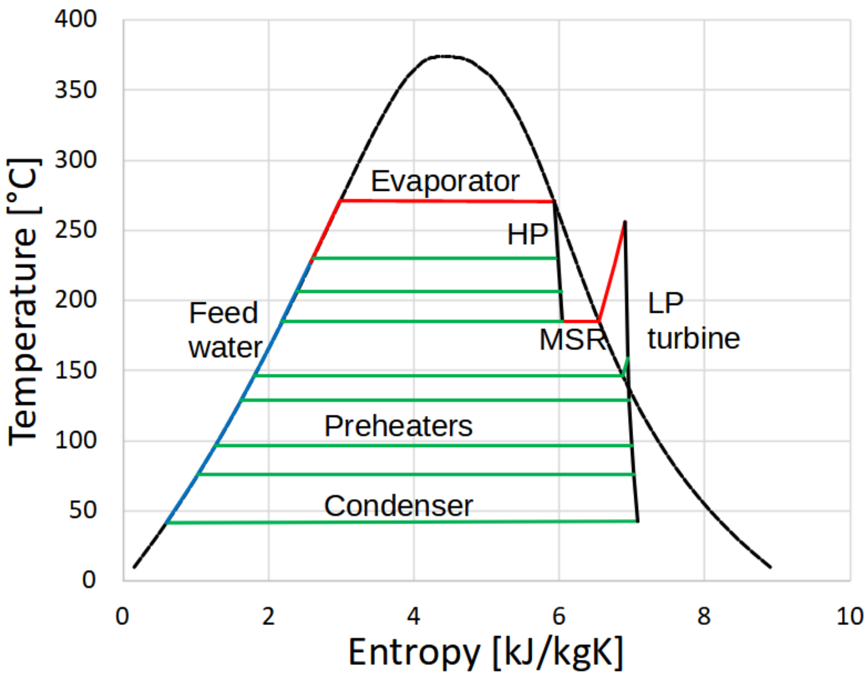

Fig. 5: Temperature-entropy diagram of the reference steam cycle.

For the SPWR, we multiplied exemplarily all pressures of the steam cycle by 2, except the condenser pressure, increased the live steam temperature to 350°C, which is 20°C less than the core outlet temperature, and the reheat temperature to 300°C, which is 19°C less than the saturation temperature in the steam generator. Running the MATLAB code again, we got a steam cycle as shown in Fig. 6. The T-s-diagram, Fig. 7, shows that the steam is now superheated by 31°C at the inlet of the HP-turbine. The net efficiency of this steam cycle is determined as 38.9 %, which is 9 % better than the reference case. Higher efficiencies can be obtained, of course, with colder cooling water or with better component efficiencies, but this would improve both cases similarly.

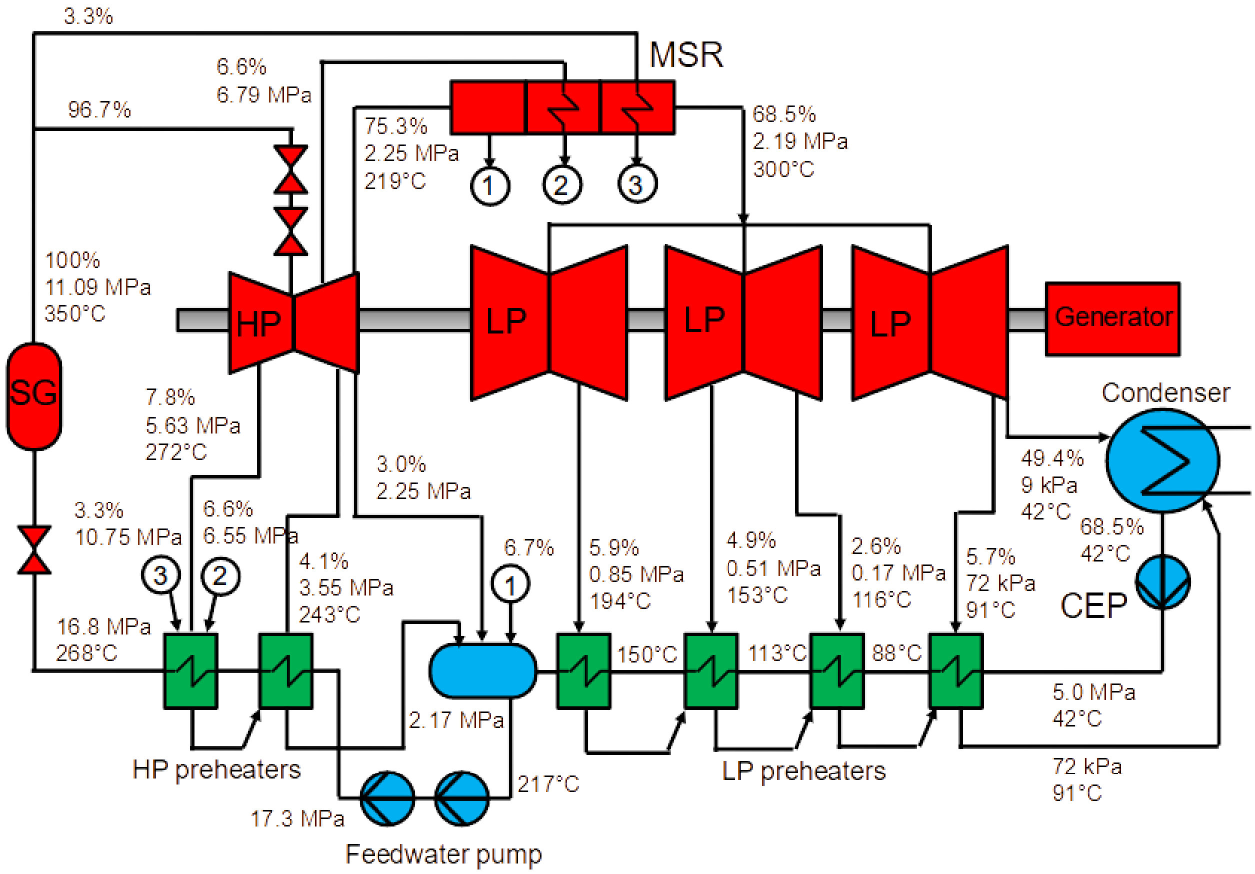

Fig. 6: Example of a secondary system for the SPWR.

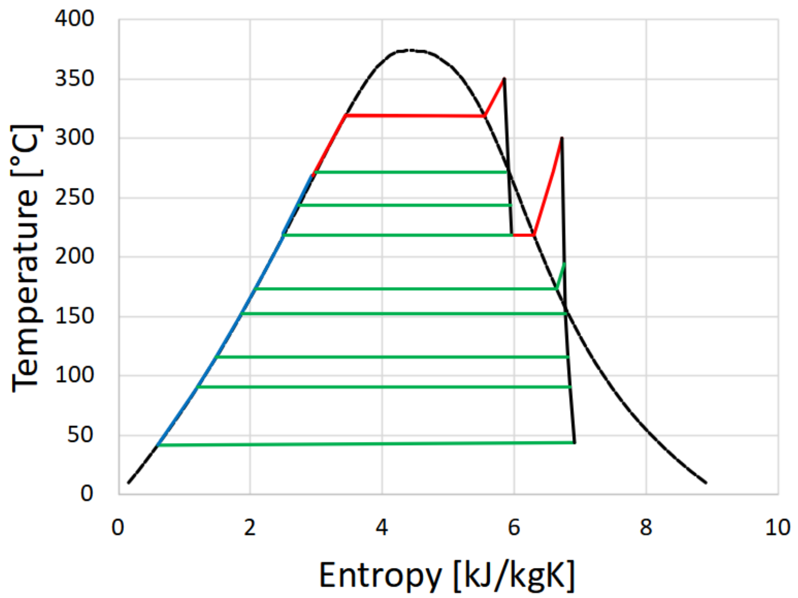

Fig. 7: Temperature-entropy diagram of a steam cycle for the SPWR.

So far, the absolute value of the mass flow has not been fixed yet. It will depend on the maximum size, which is achievable for the core, the steam generators and the steam turbines. Taking the steam mass flow of the EPR of 2556 kg/s, for which we know that LP steam turbines can be built, we need a coolant mass flow of 21,208 kg/s in the primary system, resulting in a net power of 1700 MW_e. Taking a core mass flow of 22,235 kg/s, like in the EPR, would bring the net power even up to 1780 MW_e. The target performance data of the first case are summarized in Tab. 1.

Tab. 1: Target performance data of the SPWR.

Reactor core			**Secondary system**		
Thermal power	4369	MW	Steam mass flow	2556	kg/s
Coolant mass flow	21,208	kg/s	Live steam temp.	350	°C
Inlet temperature	345	°C	Live steam pressure	11.1	MPa
Outlet temperature	370	°C	Saturation temperature	319	°C
Inlet pressure	25	MPa	Reheat temperature	300	°C
			Reheat pressure	2.2	MPa
Reactor coolant pumps			Feedwater temp.	268	°C
Inlet pressure	24	MPa	Turbine power	1803	MW
Volume flow rate	33	m^3/s	Cond. pump power	10	MW
Total power	39	MW	Feedwater pump power	54	MW
			Power plant		
			Net power	1700	MW
			Net efficiency	38.9	%

Under these constraints, each of the four once-through steam generators would need 9500 tubes of 16 m length with 16 mm outer diameter, having a wall thickness of about 1.5 mm and a pitch to diameter ratio of 1.35. The expected temperatures are shown in Fig. 8. Again, the first 1.7 m of tube length will serve as the economizer. Dryout will occur earlier at a height of about 5.2 m, where about 60 % of the feedwater has been evaporated. A detailed thermal and mechanical analysis will be needed, however, to optimize the axial position of the steam outlet.

5. Conclusions

The expected improvements of power and efficiency of the SPWR are motivating us to study the concept in more detail. Of course, the efficiency is not yet as good as predicted for a more innovative SCWR with 500°C core outlet temperature, and the concept is rather an incremental step towards it. It is not the intent of this paper to publish already first results, but rather to define a reasonable design target, supported by some rough calculations. These data can be broken down now into design requirements and constraints of major components. Each of the components will need to be dimensioned according to nuclear standards and analysed with validated codes. The general strategy is to start from proven design of PWRs, e.g. of the 3rd generation, and to increase pressure and temperature as needed, while keeping as many components from the previous design as possible. This will not only minimize technical risks, but also the engineering effort to develop it.

This design concept is offering plenty of opportunities for students of nuclear engineering to qualify themselves for a job in the nuclear industry by working out a master thesis on any of the components.

The research subject is close enough to real power plant design, such that students are motivated to take a closer look on the existing design in detail, but still challenging to exceed the former limits. Students will learn the latest methodologies of nuclear engineering, and they can validate them by comparison with the existing PWR design before they apply them to the new SPWR components.

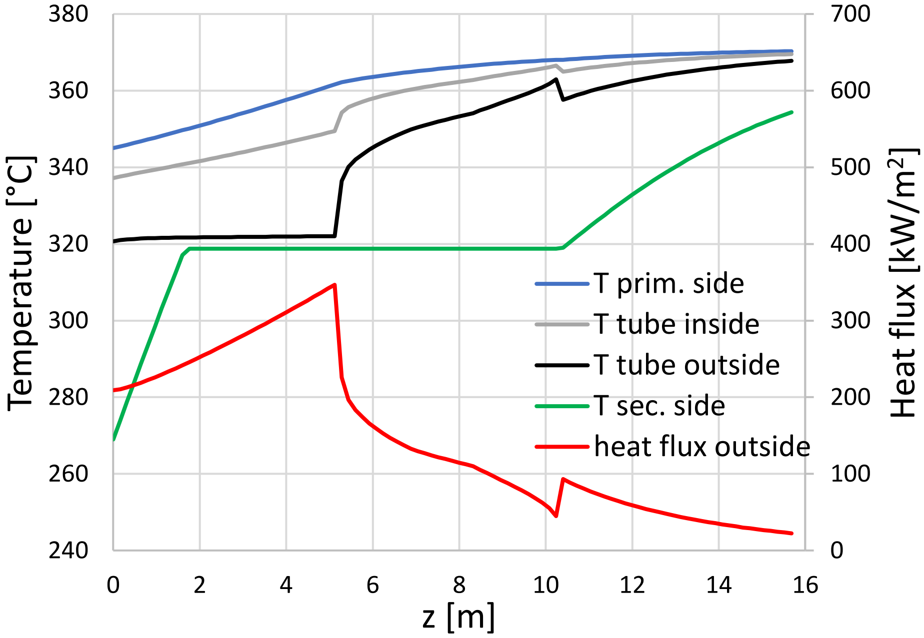

Fig. 8: : Expected temperatures of a once-through steam generator of the SPWR.

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