Design Options For High Performance Light Water Reactors

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Abstract

Light water reactors operating under supercritical pressure conditions have been selected as one of the promising future reactor concepts to be studied by the Generation IV International Forum. A first option considered in this context is a pressurized water reactor with 380°C core exit temperature, having a closed primary loop and achieving 2% pts. higher net efficiency and 24% higher specific turbine power than latest pressurized water reactors. A higher advantage can be gained from core exit temperatures around 500°C which require a multi step heat up process in the core with intermediate coolant mixing, but which achieve 44% net efficiency or more. The paper summarizes recent design approaches for such High Performance Light Water Reactors.

1. Introduction

Light water reactors, in particular pressurized water reactors (PWR) and boiling water reactors (BWR) have been among the most successful nuclear reactors during the last 40 years. More than 270 PWR have been built up to now, of which the latest ones reach a net electric power output of 1600MWe and a net efficiency of 36%. With 93 units built, the BWR had almost been as successful, even though power and efficiency levels were somewhat lower. Both reactor types are using a saturated steam cycle of around 7MPa live steam pressure, corresponding with a boiling temperature of 286°C. These live steam conditions, however, are still almost the same as those used in the 1960ies and little improvements in cycle efficiency had primarily been due to better steam turbine blades only. On the other hand, fossil fired power plants (FFPP) have increased their efficiencies significantly since then. Steam has been superheated, and live steam temperatures and pressures have been increased stepwise to 600°C and 30MPa, respectively. Since around 1990, all new coal fired power plants have been using supercritical steam conditions, reaching more than 46% net efficiency today. As a consequence, the application of such steam cycle technologies to light water reactors could offer a huge potential for further improvements.

The general advantages of such High Performance Light Water Reactors (HPLWR) are first of all a higher steam enthalpy at the turbine inlet, which does not only increase efficiency and thus reduces fuel costs, but also reduces the steam mass flow rate needed for a target turbine power. This lower steam mass flow rate reduces the turbine size as well as the size of condensers, pumps, pre-heaters, tanks and pipes, and thus the costs of the overall steam cycle. As the capital costs of nuclear power plants are usually higher than their fuel costs, this latter advantage has even a higher impact on electricity production costs than efficiency. Even more cost advantages are expected from plant simplifications such as missing steam separators or

primary pumps in case of a direct steam cycle at supercritical pressures. Another advantage of using supercritical water in a nuclear reactor is that a boiling crisis will physically be excluded, which adds a new safety feature to this design.
On the other hand, supercritical water introduces some new challenges compared with a conventional PWR or BWR. Aiming at steam temperatures similar to those achieved in FFPP, density differences of the coolant in the core will exceed those of a BWR, resulting in a lack of moderator if not compensated by other means. The hotter coolant will result in hotter cladding temperatures of fuel pins, such that stainless steel or even high temperature alloys will be needed instead of Zircalloy. Finally, the enthalpy rise in a boiler of a FFPP is more than ten times higher than the enthalpy rise in a PWR or BWR. A more sophisticated core design will be needed to avoid hot spots caused by a non-uniform power profile or by uncertainties and allowances for operation.

The following chapters will summarize different design concepts which have been developed up to now to meet these challenges. Even though fast and thermal reactor concepts have been studied, we will only address reactors with a thermal neutron spectrum within the scope of this paper.

2. Fuel assembly design concepts

First design studies had concentrated on a suitable assembly design to optimize the axial power profile under the constraint of larger coolant density differences in the core. Based on some early core design concepts studied since the 1960ies, significant new design work has been performed in the 1990ies at the University of Tokyo, initiated by Oka and Koshizuka [1]. A first fuel assembly design with additional water rods flattening the axial power profile has been presented by Dobashi et al. [2]. They used a hexagonal arrangement of fuel pins cooled by rising coolant which were housed in an assembly box like in a BWR. Additional moderator needed near the top of the core was provided by water tubes inside this assembly through which feed water was flowing preferably downwards to be mixed with additional feed water at the core bottom. These water tubes were thermally insulated to minimize heat up of the moderator water by the hotter coolant. Control rods could be inserted from the top into these water tubes. Squarer et al. [3] used this assembly design concept as a reference for their first HPLWR concept, but concluded that the radial power distribution was still rather non-uniform, requiring different enrichments in several fuel pins to homogenize the coolant heat up. A square fuel pin arrangement with up to 36 additional square water tubes, providing the moderator for the upper part of the core, could solve this issue. It has been presented by Yamaji et al. [4] and served as an option also for HPLWR core design studies by Cheng et al. [5]. Buongiorno [6] tried to avoid the additional complexity of water rods inside the assembly by selecting smaller, hexagonal assemblies with 19 fuel pins each, which were moderated by water in gaps between the assembly boxes. The inner fuel pin, however, turned out to be under-moderated again. Hofmeister et al. [7] tried to combine and optimize these concepts in a recent design study performed under the constraints that each fuel pin should be next to any moderator water, the moderator to fuel ratio should be close to a PWR to optimize the power density, and the ratio of structural material to fuel should be minimum to minimize fuel enrichment. Their result was a square arrangement with 40 fuel pins and with a single water tube in each assembly, shown in Fig. 1. Control rods were assumed to be inserted from the core top like in a PWR. As such small assemblies would require a large number of individual control rod drives, they combined 9 of these assemblies to an assembly cluster with a common head and foot piece, which will also ease handling during revisions.

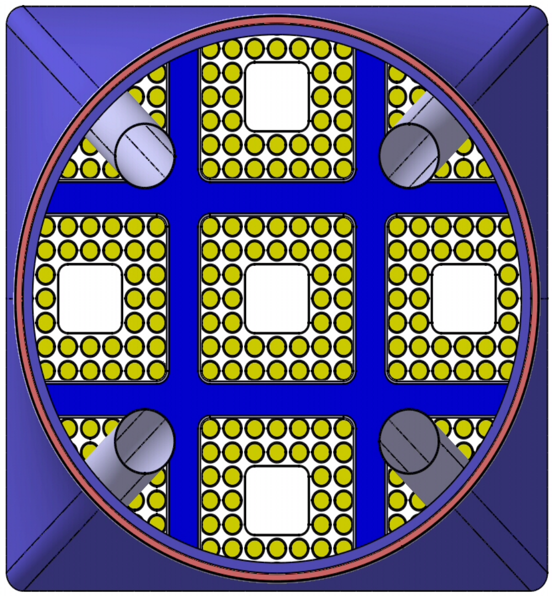

Figure 1: View through the head piece onto the HPLWR fuel assembly cluster [7].

The maximum cladding surface temperature to be allowed for operation will depend on the cladding material selected. First tests with Zircalloy indicated that rather other alloys with higher creep strength and with better corrosion resistance will be required to reach coolant temperatures beyond 500°C. Stainless steels such as SS316 or 1.4970 which were used for sodium cooled reactors in the past would be suitable candidates with sufficient creep strength up to 620°C and acceptable neutron embrittlement, as summarized by Ehrlich et al. [8]. Stainless steels and ferritic-martensitic steels were successfully tested with respect to their corrosion resistance under supercritical water conditions, as summarized by Was and Allen [9]. Recent stress corrosion crack (SCC) data by Was et al. [10], however, still left some concern at 550°C which will require further test beyond 600°C. Inconel such as IN690 or IN625 showed a smaller crack depth, but irradiated IN690 showed recently even larger SCC cracks than SS316 as reported by Teysseyre et al. [11]. This fact and the higher neutron absorption cross sections are making Inconel less favourable than stainless steels. There is some hope that oxide dispersed strengthened (ODS) materials might enable to design for cladding temperatures beyond 620°C, but further material tests will be necessary to verify this assumption.

The higher enthalpy rise of the coolant would not matter if it were uniform in the entire core. This, however, can never be guaranteed. Fuel composition and distribution, water density distribution, size and distribution of sub-channels, neutron leakage and reflector effects, burn-up effects, effects of control rod positioning or effects due to the use of burnable poisons will influence the radial power profile of the core. Material uncertainties, fluid properties, uncertainties of the neutron physical modelling, heat transfer uncertainties, uncertainties of the thermal-hydraulic modelling, scattering of the inlet temperature distribution, manufacturing tolerances, deformations during operation, or measurement uncertainties will cause a statistical scatter of the enthalpy rise. Finally, some small but allowable transients might be caused by control of power, coolant mass flow, core exit temperature and pressure. Schulenberg et al. [12] estimate that a total hot channel factor of 2 should be multiplied with the average enthalpy rise, to yield the maximum, local enthalpy rise under worst case conditions. An analogue problem is also known from FFPP boiler design. It has been solved there by splitting the total enthalpy rise into an evaporator and two successive superheaters, and by mixing the coolant homogeneously between each of these components. Next, we will show some examples of how the HPLWR design can accomplish this issue.

3. Reactor design concepts

Differences in HPLWR design concepts under discussion today are primarily due to the core design needed to achieve a certain core exit temperature. We differ between a single, two, or three pass core concept, depending on the change of flow direction during heat up of the coolant. The systematic is sketched in Fig. 2.

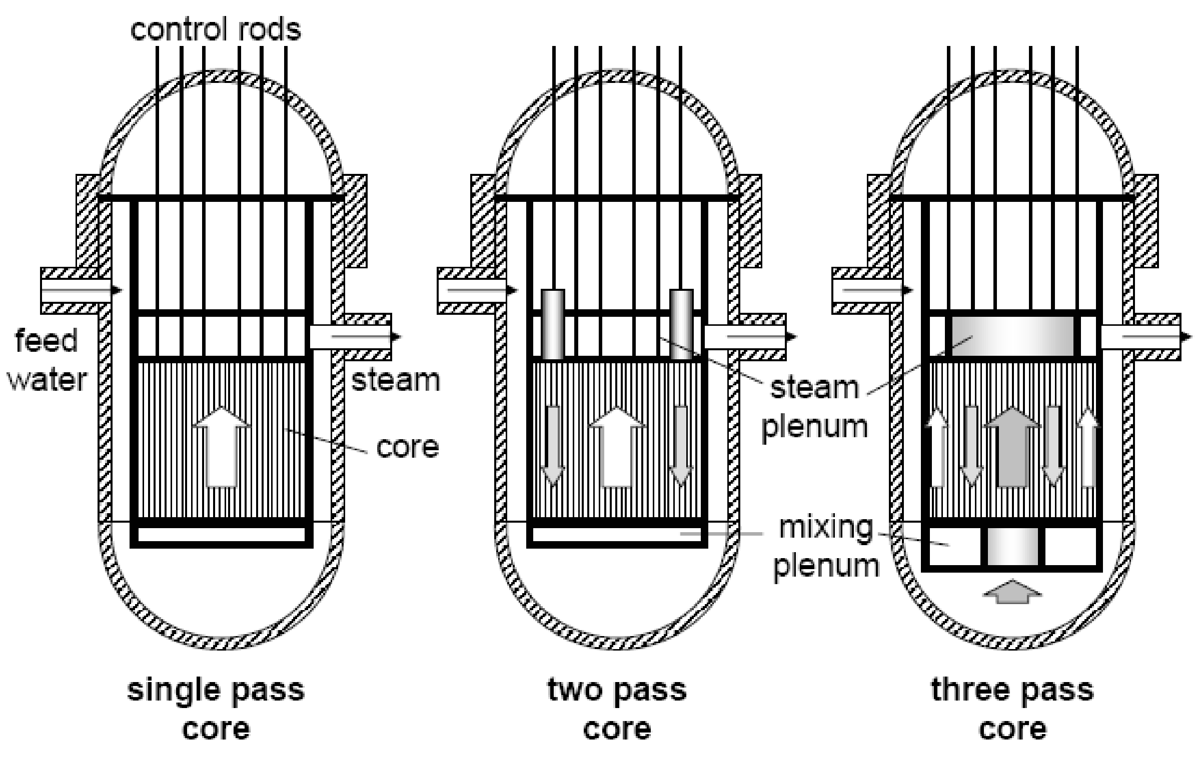

Figure 2: Core design concepts with multiple heat-up steps

The single pas core concept assumes a feed water supply at the core bottom and a hot coolant release at the top like in a PWR. If the coolant is pre-heated by downward flow in some assemblies, the mixing plenum below the core can be used to mix coolant non-uniformities before the second heat-up step with upward flow, which serves as a superheater then. We will be closer to FFPP boiler design if we heat up the coolant in three steps, starting with an evaporator with upward flow, a first mixing in a steam plenum above the core, a second heat up in a superheater with downward flow, and a third step with upward flow again after mixing in a second mixing plenum below the core. The following chapters shall illustrate some design studies taken up to now using these different concepts.

3.1 Single pass core concept

Starting from a conventional PWR design with 15MPa pressure and 325°C core exit temperature, the system pressure could be increased to 25 MPa and, in a first step, the core exit temperature to 380°C. Now, the coolant pressure will be higher than the critical pressure so that a boiling crisis will be avoided. The particular selection of the core exit temperature can better be understood when we look at the specific heat. It has a pronounced peak at 384°C, which we call the pseudo-critical temperature at this pressure. Similar to the boiling phenomenon in a PWR, this peak enables to run a sub-channel of the coolant at a significantly higher exit enthalpy, while reaching only slightly higher exit temperatures, if the average exit temperature is chosen slightly below the peak. This phenomenon is sometimes called “pseudo-boiling” of supercritical water.

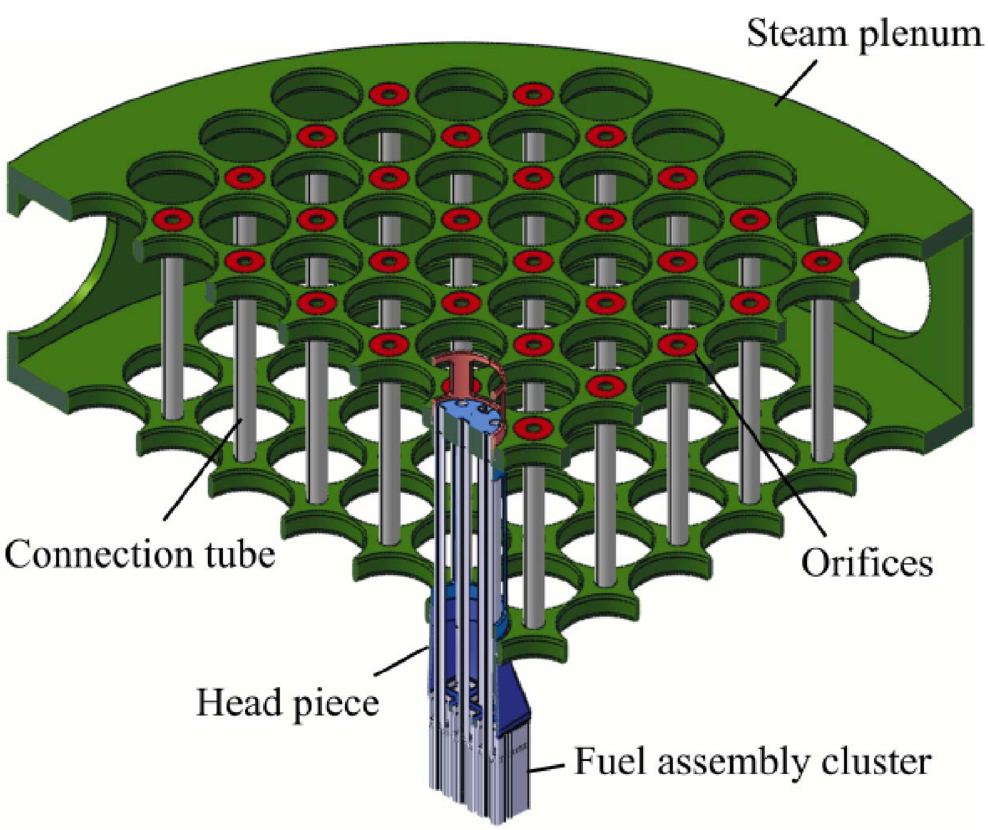

Figure 3: Steam plenum and head piece of a single pass core design [7].

Vogt et al. [13] worked out such a core design as a near term application for supercritical water technologies, called PWR-SC. They used the assembly design of Hofmeister et al. [7] with clusters of 9 assemblies with 40 fuel pins each. Moderator water is flowing downwards though water rods inside the assemblies and through the gaps between the assembly boxes, whereas the coolant rises upwards. Hofmeister et al. [7] designed a head piece for their assembly clusters which permits a counter-current flow of moderator water and core exit steam with almost no leakage, shown in Fig. 3. A transition piece reduces the square cross section of the cluster to a smaller, round cross section above, which carries 2 sealing rings. Outlet windows release the hot steam horizontally into a steam plenum mounted over all head pieces. The water tubes are welded into the top of the head piece to supply moderator water for the assemblies. The steam plenum includes a number of connection tubes which serve as stiffeners connecting the upper with the lower half of the steam plenum. They supply as well moderator water needed in the gaps between the assembly boxes.

The foot piece has been optimized such that it mixes homogenously the moderator water released from moderator tubes and from the assembly gaps with additional feed water, supplied through the downcomer directly to the core inlet. Inlet orifices underneath the foot piece adjust the coolant mass flows to the assembly power at each individual position, as shown in Fig. 4.

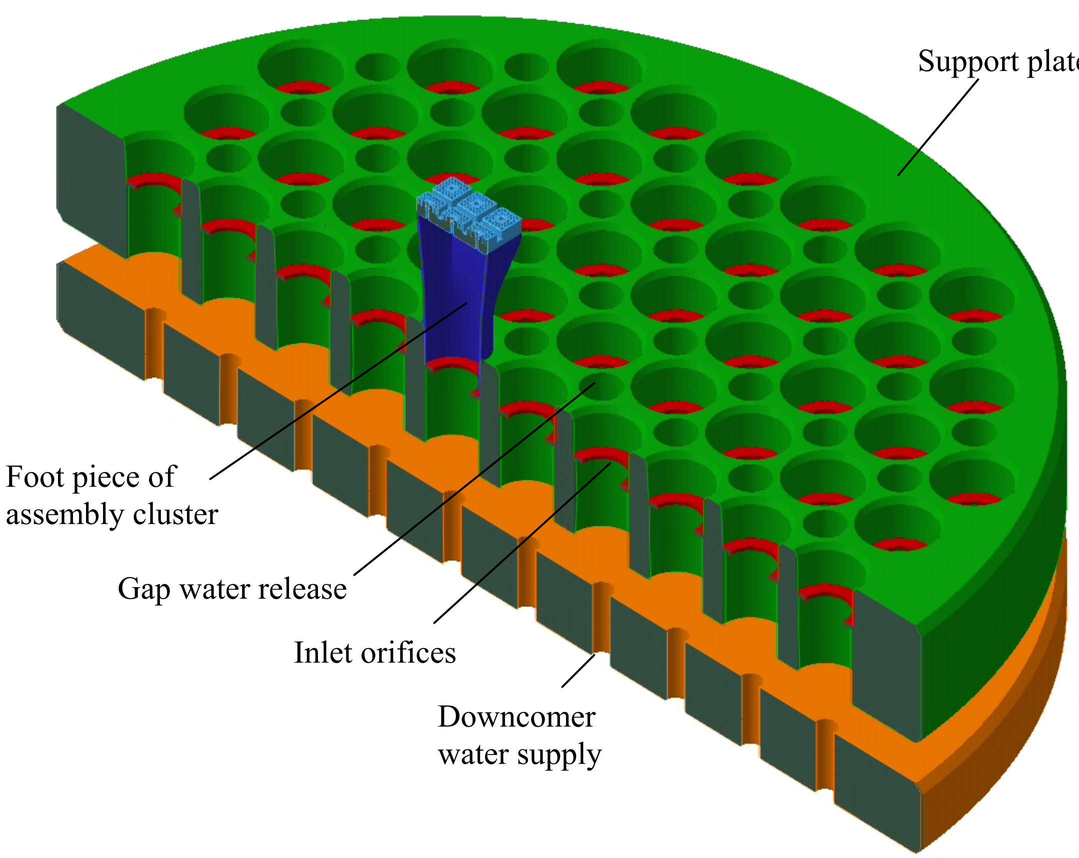

Figure 4: Foot piece and support plate with orifices of a single pass core design [18].

A reactor pressure vessel (RPV) design, shown in Fig. 5, has been worked out by Fischer et al. [14]. Like in a PWR, the core barrel is suspended in the flange of the reactor pressure vessel, and aligned only horizontally in the lower plenum. The steam plenum, however, is aligned at the outlet flanges to minimize leakage of colder feed water into the steam. Four steam outlet tubes can be driven into the steam plenum to fix it at this position. They serve also as coaxial heat shields to avoid contact of hot steam with the reactor pressure vessel and thus minimize the thermal deformations of the RPV at its flange. When the assemblies heat up, the head pieces of the assembly clusters move vertically upwards into the steam plenum, while the sealing rings still minimize leakage there. Control rods are foreseen in 5 of the 9 assemblies of each cluster. Above the core, the control rods are aligned in tubes like in a PWR.

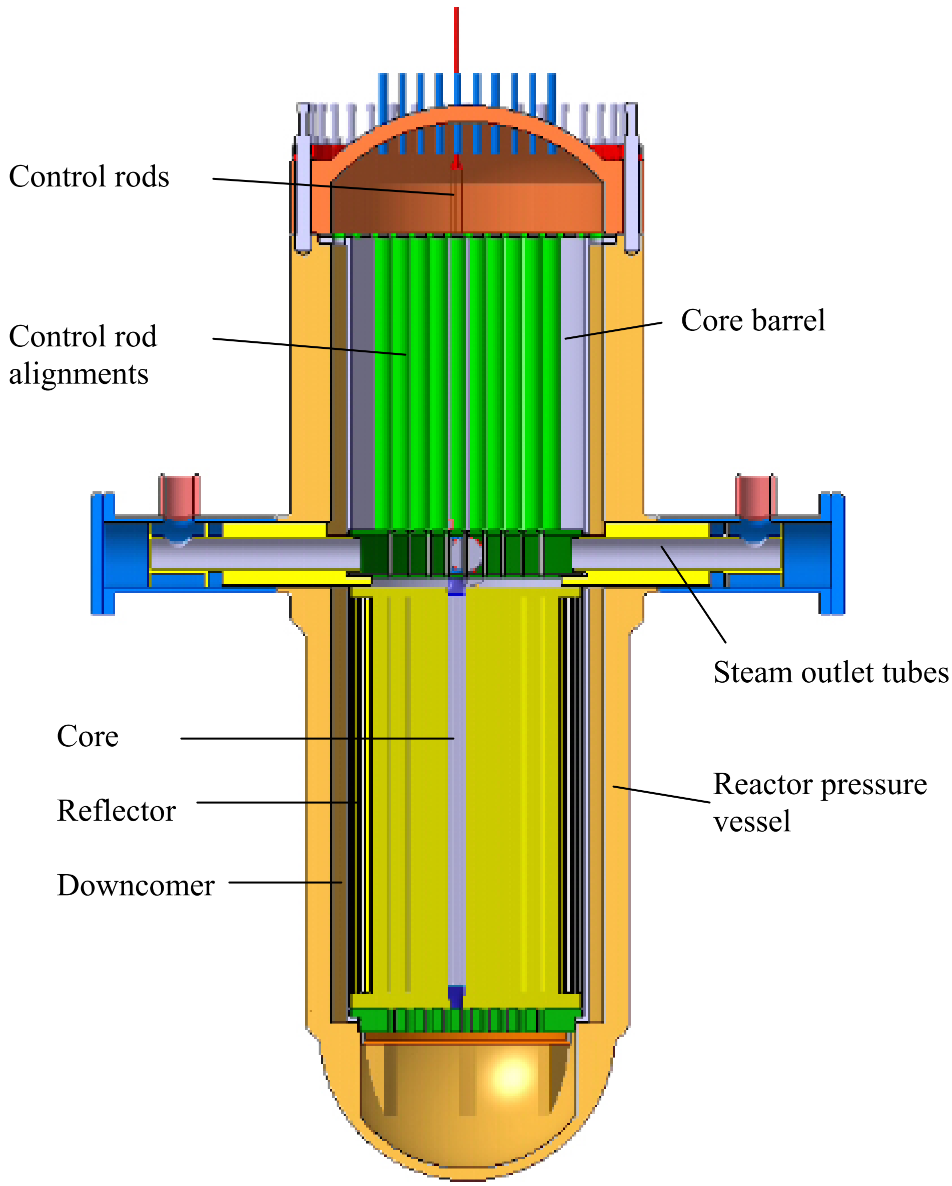

Figure 5: Reactor pressure vessel design [14].

The core designed by Vogt et al. [13] has a power density of $100MW/m^3$, which is comparable with a PWR. The maximum core exit temperature, even using a total hot channel factor of 2.14 for the hottest sub-channel under worst case conditions, reaches 416°C only. As supercritical water at a core exit temperature of 380°C is still liquid, steam generators need to be foreseen with superheaters instead of separators, which produce superheated steam at 370°C and 7.5MPa. As a consequence of the higher enthalpy rise in the core, the primary pumps will require only 24% power compared with a PWR and cross sections of the primary loop may be reduced to ¼. The higher live steam temperature of the secondary side reduces the mass flow rate there to 82% but increases the gross electric power by 3% at the same thermal power of the core. The net efficiency is predicted to be 2% points higher than the latest PWR design.

3.2 Two pass core concept

If the core exit temperature will be further increased beyond the pseudo-critical temperature, steam will be generated in the core and steam generators can be omitted. Like in a BWR, the outlet lines of the reactor may be connected directly with the inlet of the high pressure turbine which reduces the plant erection costs. The advantage of pseudo boiling, however, which keeps the core exit temperature limited, will be given up. Like in a FFPP, the coolant must now be heated up in steps with intermediate mixing to avoid overheated fuel pins in hot channels. A core design with a two-step heat up has been proposed by Kamei et al. [15], called Super LWR. They used 42% of the reactor inlet mass flow rate (at 280°C) as coolant running downwards in peripheral fuel assemblies, 50% as moderator water flowing downwards in moderator rods, and the remaining 8% are supplied through the downcomer to the lower plenum where all mass flows are added and mixed. This way, the coolant will be preheated already to around 380°C in the lower plenum. From there, it rises in the inner fuel assemblies of the core where it is heated up to 530°C average core exit temperature according to their proposal. Now, the inner fuel assemblies serve as a superheater of the core described above. The average power density of such a core, using the assembly design by Yamaji et al. [16], is $62MW/m^3$.The peak cladding surface temperature of the inner assemblies has been predicted by Yamaji et al. [16] at 732°C, not yet including uncertainties and allowances for operation. While these peak temperatures will exceed the limits of available cladding materials, a less ambitious core exit temperature of around 430°C could certainly match the creep and corrosion limits of stainless steel.

Using the assembly design of Hofmeister et al. [7], there are only a few modifications needed to enable such a flow path. While the foot piece will be unchanged, the head piece must be extended by around 10cm above the upper plate of the steam plenum, as shown in Fig. 6. Two different bushings, screwed around the head piece and sealed with C-rings, will be needed. In case that the coolant should flow downwards, a bushing with inlet openings for feedwater in combination with a long cylindrical can is screwed onto the head piece. The cylindrical can is used to close the exit windows for the steam. The size of the inlet openings in the bushing may adjust the individual mass flow rate needed for this cluster. In other cases, when coolant should flow upwards to be released to the steam plenum, the cylindrical can is removed and a closed bushing must be screwed onto the head piece instead. With such an assembly design, any position in the core can be selected for downward flow, not only the peripheral ones. Each assembly can be used either for upward or for downward flow, and the flow direction of a cluster can simply be changed during a revision by exchanging the can and the bushing. The reactor pressure vessel design can still be used without any major change.

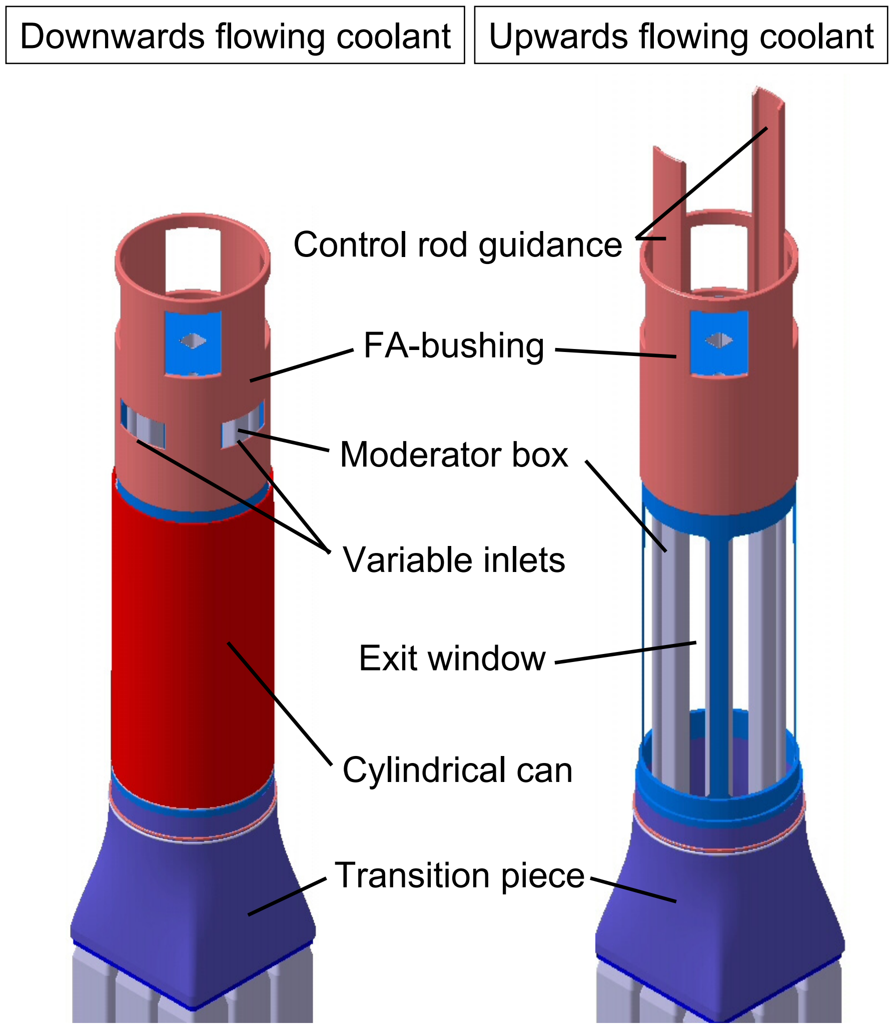

Fig. 6 Modified head piece to enable upward and downward flow

3.3 Three pass core concept

Higher core exit temperatures, and thus a higher specific turbine power and a higher net efficiency, can be achieved if we strictly follow the concept of supercritical FFPP and include even a second superheater. The resulting three pass core concept has been sketched by Schulenberg et al. [12]. An evaporator (or rather pseudo evaporator in case of supercritical water) is situated in the centre of the core. Underneath its inlet at the core bottom, all moderator mass flows from moderator tubes and from gaps between assemblies are mixed with feedwater from the downcomer to an inlet temperature of around 310°C. The evaporator heats up the coolant to 390°C. An inner mixing chamber above the core eliminates hot streaks. A first superheater with downward flow, surrounding the evaporator, heats the coolant up to 433°C. After a second mixing in an outer mixing chamber below the core, the coolant will finally be heated up to 500°C with upward flow in a second superheater in peripheral assemblies. Schulenberg et al. [12] could show with a simplified single channel analysis that peak cladding temperatures of around 620 – 630°C can be expected in all heat up steps, even if a hot channel factor of 2 is assumed to scale a hot coolant sub-channel from an average sub-channel.

A mechanical design for such a flow path has not yet been worked out. Starting from the single pass core concept, we will need a modification of the foot piece of the assembly cluster to avoid a release of colder moderator water into the hot, superheated steam in the outer mixing chamber below the core. Moreover, the upper and lower mixing chambers will differ significantly from the single path design as an inner chamber with a lower temperature needs to be separated from an outer one. The European consortium of the project "HPLWR Phase 2", which started recently in September 2006, agreed to work on this concept with the objective to access its feasibility, its safety features and its economic potential. Schulenberg and Starflinger [17] report about more details of this project.

4. Conclusions

Light water reactor technologies of PWR and BWR and advanced steam cycle technologies of fossil fired power plants have been well proven within the last 4 decades. Both are among the most reliable power plant technologies available today. Combining both technologies shall lead to a Generation IV nuclear reactor concept achieving a higher specific turbine power and a higher thermal efficiency than conventional light water reactors. Three examples of core concepts, described above, illustrate that the complexity of the core design will increase with increasing core exit temperature. Nevertheless, most technologies from PWR and BWR can still be used, such as the PWR reactor pressure vessel design including its control rod drives, the PWR or BWR safety system or the containment design. The advanced steam cycle, on the other hand, can be adopted from fossil fired power plants, such as the turbine design, valve and pump design, and the feedwater system. The consequent transfer of FFPP technologies to conventional light water reactors is considered to be the key for success of the development plan.

5. Acknowledgement

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