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The Simplified Supercritical Water-Cooled Reactor (SSCWR), a new SCWR design

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A B S T R A C T

The Supercritical Water Cooled Reactor (SCWR) was chosen as one of the Generation IV reactors by GIF. At the moment, a number of concepts exist, such as the American SCWR, the Canadian CANDU–SCWR, the European HPLWR (High Performance Light Water Reactor), the Japanese Super LWR and the Korean SCWR. The driving force behind the developments is the fact that the fossil fired power plants are continuously increasing their efficiency by use of higher pressures and temperatures.

The SCWR has numerous advantages over today's wide-spread Light Water Reactors. On the other hand, the scientific community must cope with various challenges before the first power plants of this type can be built. The European and the Japanese concepts are the most promising, although the complicated water flow path repeatedly sets newer challenges. Simplicity is always a good choice in reactor design, thus the authors went back to the basic once-through cycle, significantly improving it. With the combination of zirconium-hydride as extra moderator, axially varying fuel enrichment and moderation, longer active length and smaller core diameter (resulting in thinner walls for the reactor pressure vessel), a new and simpler design is proposed in this paper. This design is called Simplified Supercritical Water-Cooled Reactor (SSCWR) and adapts better for the different operating conditions and burn-up, therefore it is inherently safer and more reliable.

1. Introduction

Supercritical Water Cooled Reactors (SCWRs) were already investigated in the '50s and '60s. A good overview is given by Oka (2000). This topic was picked up again by the University of Tokyo in the '90s and by the Russian Kurchatov Institute which designed the B-500 SKDI (Slin et al., 1993). The driving force behind the developments is the fact that the fossil fired power plants are continuously increasing their efficiency by use of higher pressures and temperatures. Scientists from the University of Tokyo showed that similar efficiencies are also achievable in the case of nuclear power plants (Oka and Koshizuka, 1998). These results encouraged other countries to develop their own concepts:

- the HPLWR project was launched in 2000 with support from the European Union and is now being followed by the HPLWR Phase 2 project (started in 2006, with a duration of 42 months);
- the USA made a feasibility study of the SCWR between 2001 and 2004;
- Korea started their research in 2002;
- in the case of CANDU, developing a supercritical pressure version is obvious (also called CANDU–SCWR).

The SCWR has many advantages over today's wide-spread Light Water Reactors. On the other hand numerous challenges must be solved before the first power plants of this type can be built. Of these challenges we would like to emphasize two: (1) cladding materials have to withstand high temperatures (600–650 °C in normal operating conditions), neutron irradiation and should be as transparent as possible for neutrons; and (2) heat transfer phenomena must be understood thoroughly.

Although SCWRs can be designed also as fast reactors, in this article we focus on the thermal concepts. The European HPLWR and the Japanese Super LWR concepts seem the most promising. They use water rods as extra moderator to overcome the sharp decrease (also called pseudocritical transformation) in the density of the coolant after the pseudocritical point, and multi-pass cores to fulfil the design criteria of 620 °C for the cladding temperature. On the other hand, these innovations result in a complicated water flow path in the reactor pressure vessel. In steady-state calculations, an optimum of the flow distribution is assumed and other distributions such as power and temperature are calculated accordingly for all regions of the core. However, these distributions are only valid for nominal conditions. It is obvious that in the case of transients and

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during burn-up the power distribution changes, which may shift the temperature distribution into unfavourable states, e.g. the pseudocritical transformation could shift to another region. Thus, determining the transient behaviour of multi-pass cores is much more difficult than that of single-pass cores. Another disadvantage of multi-pass cores is that they require reactor pressure vessels of greater diameter which limits the thermal and electric power of the core.

With the combination of zirconium-hydride as extra moderator, axially varying fuel enrichment and moderation, longer active length and smaller core diameter (resulting in thinner walls for the reactor pressure vessel) a new and simpler design is proposed in this paper. This design is called Simplified Supercritical Water-Cooled Reactor (SSCWR) and adapts better for the different operating conditions and burn-up, therefore it is inherently safer and more reliable.

2. Some SCWR concepts

Most thermal SCWR concepts studied so far can be generalized in the following manner. The assemblies are of square or hexagonal type with water rods or solid pins (mostly zirconium-hydride) as extra moderator. The high enthalpy rise in the core forced the scientists to use longer flow paths which resulted in the so called two- and three-pass cores. They have the advantage of reducing the maximum cladding temperature; on the other hand, the simplicity of a one-pass core is lost. The four main concepts are as follows.

2.1. Japanese concept

One of the first concepts (Oka et al., 1992) featured a core with hexagonal assemblies and extra moderator in the form of zirconium hydride (ZrH). Because of safety requirements, the outlet temperature of the coolant was limited to 400 °C. Later the ZrH rods were replaced by water rods with downwards flow, the cladding was modified to be Ni-alloy, square assemblies were introduced instead of hexagonal ones and a two-pass core was adopted. This concept was named SCLWR-H (Dobashi et al., 1998; Yamaji et al., 2003) and was further optimized resulting in the design called Super LWR (Kamei et al., 2005). The major modifications which lead to the Super LWR were the use of stainless steel instead of Ni-alloys as cladding material and the use of Gd_2O_3 as burnable poison. Both resulted in a more favourable power distribution and a lower average enrichment (6.11%) for the fuel rods.

2.2. American concept

The American scientists examined numerous materials to choose the best one for the role of moderator. They found that ZrH_x has many advantages (Buongiorno and MacDonald, 2003), still they adopted a similar concept as the Japanese: square assemblies with water rods which have a constant cross section along the axial direction. The calculations showed that the water rods should be insulated which was accomplished with zirconium-oxide. The core was designed to be a 1-pass core with an outlet temperature of 500 °C (MacDonald et al., 2004).

2.3. Korean concept

The assemblies are of square-type with a cruciform type U/Zr solid moderator to avoid complicated water flow paths. The enrichment of the fuel pins varies axially, as well as radially. The height of the assemblies is 381 cm. A 1400 MW_e power plant with an outlet temperature of 510 °C was designed (Bae et al., 2007).

2.4. European concept

The European development is coordinated and mainly carried out by Forschungszentrum Karlsruhe and the concept is called High Performance Light Water Reactor (HPLWR). The starting point for the HPLWR was the Japanese design by Dobashi et al. (1998). The assembly was reconstructed by Hofmeister et al. (2005) based on extensive mechanical, neutronics and thermal-hydraulics research. In the original one-pass core they faced the problem of hot spots which arose because of local power peaking. This resulted in excessive cladding temperatures (above 630 °C), thus a three-pass core was developed (Schulenberg et al., 2006) keeping the envisioned outlet temperature of 500 °C (the outlet temperature of the one-pass core was reduced to about 380 °C and is now called PWR-SC, Vogt et al., 2006).

3. Motivation and goals

In this paper a new SCWR design, the SSCWR is proposed, which features some new ideas and also combines several from the various concepts presented before. The basic ideas behind our effort to eliminate drawbacks of the concepts presented in Section 2 and to improve the design are as follows:

(1) A one-pass core is chosen to simplify the construction of the reactor. In our view this in itself reduces the failure possibilities, further increases reliability and inherent safety, and finally adapts better to the varying operating conditions and to the changes during burn-up.
(2) The decrease in moderation due to the density drop of the coolant in its flow direction must be proportionately compensated by the increase of the moderator to fuel atomic ratio. This can be achieved in two ways, namely:
  (2a) using fuel pins with different length, i.e. increasing the water to fuel volume ratio in the flow direction;
  (2b) since the preceding countermeasure is insufficient, moreover raises other problems which are explained in detail in Section 4, it is supplemented by the use of zirconium-hydride rods as extra moderator on top of the shorter fuel pins.
(3) Axial zoning of the fuel enrichment is used, because it gives another possibility to control the moderator to fuel atomic ratio and, at the same time, it helps to achieve the desired power distribution.

Combination of the above points proved successful, moreover resulted in numerous favourable effects, e.g.:

- The diameter of the reactor pressure vessel can be decreased by means of greatly reduced cross section of the core. Thus, the walls of the vessel become thinner, which calls forth easier manufacturing.
- The distance between the core and the reactor pressure vessel can be increased by means of a larger downcomer region. This in turn leads to reduced radiation damage of the vessel through.

The article contains the results and analysis of detailed calculations of the SSCWR. The main criterion followed by the authors is to achieve an outlet temperature of 500 °C with a one-pass core arrangement (inlet temperature is 280 °C, system pressure is 25 MPa, mass flow rate is 1179.0 kg/s, thermal power is about 2300 MW[1]).

[1] These values are adopted from the HPLWR.