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Review of R&D for supercritical water cooled reactors

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ABSTRACT

The Supercritical Water-Cooled Reactor (SCWR) is a high temperature, high pressure water-cooled reactor that operates above the thermodynamic critical point (374 °C, 22.1 MPa) of water. In general terms, the conceptual designs of SCWRs can be grouped into two main categories: pressure vessel concepts proposed first by Japan and more recently by a Euratom partnership, and pressure tube concepts proposed by Canada, generically called the Canadian SCWR. Other than the specifics of the core design, these concepts have many similar features, like outlet pressure and temperatures, steam cycle options, materials, or heat transfer characteristics. Therefore, the R&D needs for each reactor type are common, which enables collaborative research to be pursued. The paper provides an overview on research and development performed so far on the SCWR within the Generation IV International Forum.

1. Introduction

Looking at the trend of coal fired power plants in the last 40 years, we observe a remarkable increase of net efficiency from around 37% in the 1970s to more than 46% today. The last 20 years since 1990, in particular, were characterized by an increase of live steam temperature beyond 550 °C, when boiler steels became available which allowed exceeding the former material limits. Along with the temperature increase, the live steam pressure went up to maximize the turbine power, finally exceeding the critical pressure of water. The next generation of coal fired power plant will even reach a net efficiency of ~50%, when live steam temperatures of 700 °C or more can be realized. In comparison with such development, the net efficiency of latest pressurized water reactors (PWR) of around 36% is still close to the efficiency of ~34% of the first generation of light water reactors.

Following this trend, Super-Critical Water-cooled Reactors (SCWRs) are a class of high temperature, high pressure, water-cooled reactors that operate above the thermodynamic critical point of water (374 °C, 22.1 MPa). The GIF Technology Roadmap (GIF, 2002) has identified several of the key technical advantages of the SCWR compared to conventional water technologies that make it attractive for consideration as a Generation IV Nuclear Reactor System. The main thrusts of these advantages translate into improved economics because of the increased thermodynamic efficiency and plant simplification opportunities afforded by the high-temperature, single-phase steam.

Other key advantages of the SCWR include:

- SCW fossil-fired plants (SCW-FFPs) are well known in the electricity-production industry and, in many cases, vendors of nuclear products are also manufacturers of components for SCW-FFPs.
- No turbine development is required for outlet temperatures <625 °C; SCW-FFPs are already operating at these conditions.
- Advanced fuel cycles can be considered, but development of a new fuel type is not essential before a prototype reactor could be built.
- The SCWR is an evolution from existing water reactors (i.e., pressurized water reactors (PWRs), boiling-water reactors (BWRs) and pressurized heavy-water reactors (PHWRs)); thus, today's nuclear expertise is relevant and can be leveraged in the development of the concept.
- Many existing utilities around the world will be comfortable with SCWR technology since they currently operate both nuclear power plants and SCW-FFPs.
- Projections of the cost of the SCWR will be more accurate than those of other systems under development, since cost models will be based in a large part on proven systems.

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2. Past experience

Supercritical water cooled reactors were studied already in the 1950s and 1960s as summarized by Oka (2000). In particular, we like to mention the following early studies:

- A water moderated, supercritical steam cooled reactor was designed by Westinghouse in 1957, in which 7 fuel rods in cylindrical, double walled cans formed the fuel assemblies to insulate the superheated steam from the liquid moderator water at 260 °C. An indirect steam cycle was favoured for this concept to avoid activity in the turbines.
- A heavy water moderated reactor, cooled with light water, was designed by General Electric in 1959 for a thermal power of 300 MW with a once through steam cycle. The coolant passed the core four times, reaching an outlet temperature of 621 °C.
- A graphite moderated and light water cooled pressure tube reactor was designed by Westinghouse in 1962, called the Supercritical Once Through Tube Reactor (SCOTT-R) for an electric power of 1000 MW with a thermal efficiency of 43.5%. The low pressure tank containing the graphite moderator was cooled with Helium.
- A pressurized water reactor with a closed loop primary system at supercritical pressures had already been proposed in 1966.

A supercritical water cooled reactor, however, has never been built in the past. Instead, a boiling water reactor with a nuclear superheater was built in Grosswelzheim, Germany, which could be considered as an early, evolutionary step from boiling water reactors towards an SCWR. The HDR (Heissdampfreaktor) was intended to reach 500 °C core outlet temperature in its final stage, and the prototype built from 1965 to 1969 with 100 MW thermal power was designed for a reduced temperature of 457 °C of superheated steam at 9 MPa reactor inlet pressure as an introductory step. Schulenberg and Starflinger (2012) summarized the key design features.

The construction of the power plant started in Jan. 1965. First criticality was reached on Oct. 14, 1969, and the HDR power plant was connected to the electricity grid on the same day. Commercial operation started on Aug. 2, 1970, reaching up to 23 MW_e, but the core was damaged soon. It has been reported that the tubes of the superheated steam collapsed, but details have not been published. The reactor tests were finished and the reactor was shut down on April 20, 1971, 18 months later, having produced 6200 MWh electric power in total. It is, therefore, not a story of success but still a milestone in the development of light water reactors with increased temperatures.

3. Applications for SCWR technologies

The basic idea of using a supercritical steam cycle for nuclear power production can be applied to a number of different systems, as will be discussed with the following examples which had been worked out by the Generation IV International Forum within the last 10–15 years.

3.1. Pressure vessel type reactors with a thermal neutron spectrum

In Europe, a consortium of 12 organizations from 8 European countries started in 2006 to address this challenge by working out a design concept of such a reactor, which is referred to as 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 core with a thermal neutron spectrum, a net electric power of 1000 MW and a net plant efficiency of around 44% for base load electric power production. Meanwhile, the conceptual design has been completed. Schulenberg and Starflinger (2012) reported about design details and supporting analyses.

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 supercritical fossil fired power plants of 280 °C, the enthalpy rise in the core would exceed the one of conventional light water reactors by almost a factor of ten. A conventional LWR 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 pressures) and a first and second superheater with higher temperatures but lower powers 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. Schulenberg and Starflinger (2012) proposed a thermal core concept in which the evaporator assemblies are placed in the centre of the core, followed by first superheater assemblies with downward flow surrounding them, and second superheater assemblies with upward flow at the core periphery where the fissile power is low anyway because of neutron leakage.

Like in pressurized water reactors, the reactor internals include the core barrel with its core support plate and the lower mixing plenum, the steel reflector, the steam plenum with adjustable outlet pipes and the control rod guide tubes. The core barrel is composed of a cylindrical part with flange and the lower core support plate with orifices as shown in Fig. 1. The circular lower mixing plenum, which is welded to the bottom of the lower core plate, homogenizes the water flow from the downcomer before it enters through the plate into the lower part of the evaporator of the core. An annular mixing chamber underneath the core support

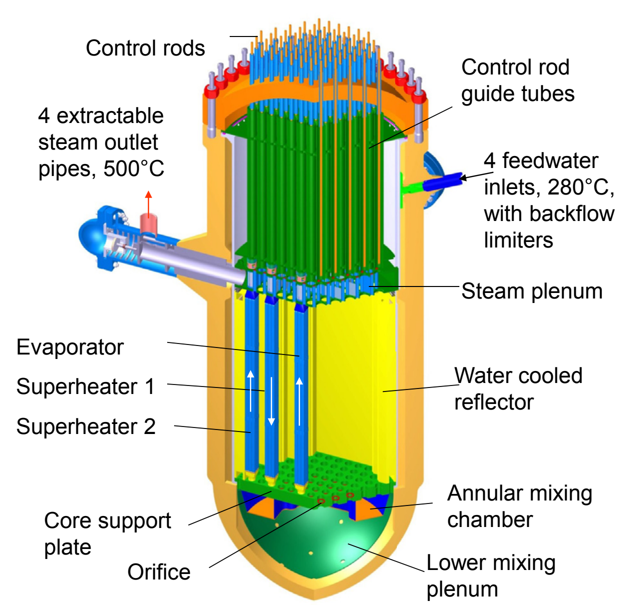

Fig. 1. HPLWR reactor pressure vessel and core structures (Schulenberg and Starflinger, 2012).