IAEA-TECDOC-1869

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Status of Research and Technology Development for Supercritical Water Cooled Reactors

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International Atomic Energy Agency

STATUS OF RESEARCH AND TECHNOLOGY DEVELOPMENT FOR SUPERCRITICAL WATER COOLED REACTORS

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STATUS OF RESEARCH AND TECHNOLOGY DEVELOPMENT FOR SUPERCRITICAL WATER COOLED REACTORS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2019

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Printed by the IAEA in Austria
April 2019

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Status of research and technology development for supercritical water cooled reactors / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2019. | Series: IAEA TECDOC series, ISSN 1011–4289 ; no. 1869 | Includes bibliographical references.
Identifiers: IAEAL 19-01233 | ISBN 978–92–0–101919–6 (paperback : alk. paper)
Subjects: LCSH: Water cooled reactors. | Nuclear reactors. | Nuclear engineering.

FOREWORD

There is considerable interest in both developing and developed countries in the design of innovative water cooled reactors (WCRs) and, owing to the higher thermal efficiency and significant system simplifications, supercritical water cooled reactors (SCWRs). Because the operating pressure and temperature are much higher in SCWRs than in conventional WCRs, and the core flow channel configuration is completely different from that of supercritical fossil fuel power plant boilers, realizing the SCWR concept will require extensive, comprehensive research and development (R&D). R&D activities on SCWRs to date have focused on three major areas: thermohydraulics, materials and chemistry. Fundamental research has led to an understanding of important phenomena in SCWR technologies and provided information required for the next step of development. Currently, a few concepts have been assessed as being technically feasible, and several other concepts are under development. These concepts are described in this publication, together with detailed analysis of remaining gaps requiring future R&D.

The IAEA started its activities on SCWRs in 2006 and launched its first project in 2008. Since then, it has coordinated several projects to facilitate collaboration on R&D and information exchange among interested institutions in Member States. With the fast pace of SCWR development, there is strong interest in obtaining up to date information on key areas of technology. In this connection, in 2016 the IAEA held two related technical meetings: the Technical Meeting on Heat Transfer, Thermal-Hydraulics and System Design for Supercritical Water Cooled Reactors, in Sheffield, United Kingdom, 22–24 August 2016; and the Technical Meeting on Materials and Chemistry for Supercritical Water Cooled Reactors, in Řež, Czech Republic, 10–14 October 2016. In total, 52 experts from 15 Member States and two international organizations participated in these two meetings. These experts presented their R&D activities and results, and exchanged information on the state of the art of SCWR technologies. They also discussed the current progress in the three main areas of R&D, and agreed on future R&D to address the remaining issues.

The present publication, which summarizes the current status of R&D, is based mainly on the presentations and discussions that took place during these two technical meetings. It provides a comprehensive, objective overview of SCWR related R&D activities for researchers, engineers, nuclear system designers, university professors, graduate students and others engaged in the development of SCWR designs.

The IAEA appreciates the contributions of the consultants who drafted and reviewed this publication and the participants in the technical meetings. The IAEA acknowledges the assistance of C. Takasugi (United States of America) in the preparation of this publication. The IAEA officers responsible for this publication were T. Jevremovic and K. Yamada of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The Supercritical Water Reactor (SCWR) is a water cooled reactor (WCR) concept that operates at supercritical pressures [1]. When water is pressurized above its thermodynamic critical pressure of 22.1 MPa, strict distinction between liquid and steam disappears. In the SCWR cores, reactor coolant is low temperature, high density water (liquid like fluid) at the inlet, and it becomes high temperature, low density fluid (steam like fluid) at the outlet, which facilitates a once-through reactor as in a pressurized water reactor (PWR) and a direct thermodynamic cycle as in boiling water reactor (BWR). The use of supercritical pressure water as coolant enables achieving higher thermal efficiency (up to 48%) and implementing simpler heat transport system configuration than those of conventional WCRs. Hence, the SCWR concepts have the potential for improved economics [2].

A typical SCWR system is designed to operate at the pressure of 25 MPa, with average core outlet coolant temperature targeted to values in the range of 500−625 °C depending on the concepts (as described in Section 2.1) [1]. FIG. 1 compares pressure and temperature ranges in a typical SCWR with conventional WCRs.

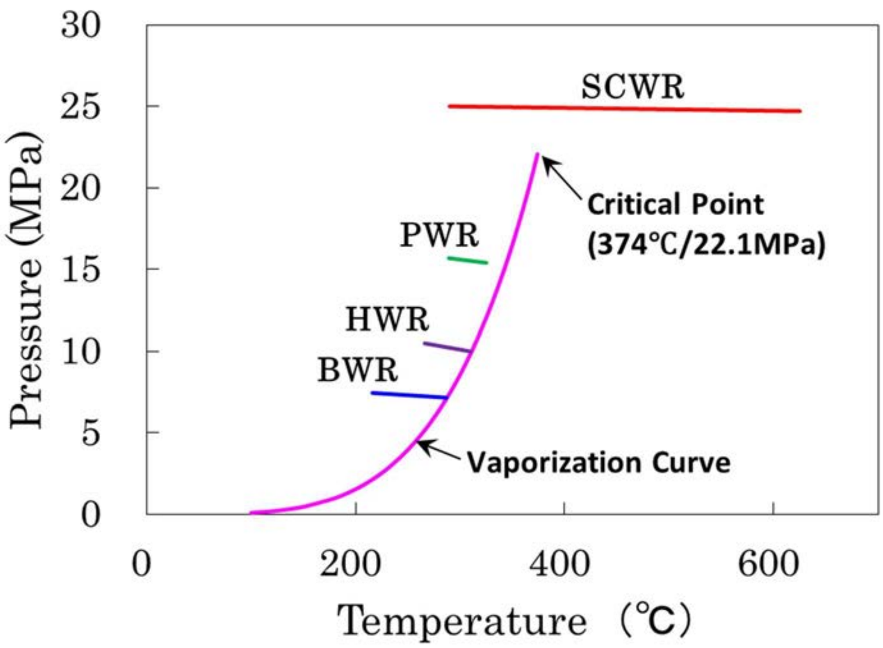


FIG. 1. Typical pressure and temperature ranges of coolant for various types of water cooled reactor (reproduced courtesy of Yamada [3]).

The outlet coolant temperature of conventional WCRs is generally limited to the saturation temperature of its operating pressure because heat transfer coefficient from fuel to coolant decreases drastically when fuel rod surfaces are covered with steam (so called 'boiling crisis'). This phenomenon is irrelevant to SCWR at normal operation because no boiling occurs at fuel rod surfaces. However, attention is required to the heat transfer 'deterioration' phenomena, which could occur at low flow rate and high heat flux conditions with a sharp increase in cladding temperature. The temperature rise is expected to be much milder when compared to 'boiling crisis'.

One of main advantages of the SCWR is that most systems are based on the existing WCR technologies, in which extensive design, construction and operating experiences have been accumulated worldwide, and on supercritical fossil fuel power plant technologies, which also come with substantial experiences in design, construction and operation. Moreover, compared to other innovative concepts (e.g., liquid metal technologies), water is chemically inert, cheap, transparent, and nontoxic, thus simplifying inspections and maintenance of this type of reactors.

Proposed SCWR design concepts are expected to incur low capital cost attributed to high thermal efficiency and simplified reactor system configuration. Without major primary circuit components, which reduce the size of the system, a SCWR design could be ideal for small modular reactors (SMRs) that are being adopted by many countries in the world [4].

Due to the flexibility of SCWR designs, the core can also be configured as fast neutron spectrum, as described in Section 2.1. The possibility of a fast neutron spectrum core would enable the SCWR to be a high converter, or even a breeder, or a burner of minor actinides.

In order to realize the SCWR concept, extensive and comprehensive R&D are necessary mainly because the operating pressure and temperature conditions are different from those of conventional WCRs, and its flow channel configuration is completely different from that of supercritical fossil fuel power plant boilers. One of the most important R&D areas is thermal hydraulics of supercritical pressure water in fuel assemblies. Precise prediction of fuel cladding temperature is indispensable to the evaluation of fuel rod integrity and safety margin. Other critical areas are the identification of materials for in-core structures, especially for fuel cladding, and the establishment of a strategy for water chemistry to minimize corrosion and activity transport. It is necessary to use materials that can withstand high pressure and high temperature water conditions combined with radiation. In addition, a development of the analysis tools for system integration and safety assessment of new reactor system designs is necessary. These R&D needs are commonly required for all SCWR concepts, and thus it makes a collaboration between institutions with R&D activities and/or developing their own design concept highly beneficial.

1.2. OBJECTIVES

Objectives of this TECDOC are to (i) provide an overview of the current R&D status relevant to SCWRs and (ii) disseminate the summarized information, based on presentations and discussions, from two IAEA technical meetings on SCWRs held in 2016 (i.e., the Second Technical Meeting on Heat Transfer, Thermal Hydraulics and System Design for SCWRs in Sheffield, UK, 22–24 August 2016; and the Third Technical Meeting on Materials and Chemistry for SCWRs in Řež, Czech Republic, 10–14 October 2016). Participants at these meetings included 52 experts from 15 Member States and two international organizations. These experts gave presentations on their R&D activities and results. They also confirmed recent progress in R&D and discussed remaining issues in the relevant areas.

1.3. SCOPE

Rapid dissemination and sharing of the latest advancements at various technology areas would expedite the SCWR development process. The IAEA has initiated activities through hosting cooperative research projects (CRPs) and organizing technical meetings to support these efforts. This would ensure technical experts and SCWR designers receiving up to date information to enhance or optimize the design. The scope of this TECDOC is to compile the state of the art information from two technical meetings for the rapid dissemination to experts in various Member States.

1.4. STRUCTURE

This TECDOC consists of two parts. The first part (Section 2) provides an overview of R&D efforts in Member States to establish SCWR concepts. It describes the R&D areas such as thermal hydraulics phenomena including heat transfer, pressure drop, critical flow, materials and chemistry including materials for fuel cladding and in-core structure and supercritical water radiolysis. The section focuses on the core and fuel concepts but provides also the system concepts for selected designs, when available. It is intended for this TECDOC to provide a comprehensive and objective overview of SCWR R&D to researchers, engineers, and nuclear system designers, university professors and graduate students, and others engaged in and/or related to the development of SCWRs. The second part (Section 3) with the attached CD presents the proceedings of these two technical meetings with the goal to disseminate information on recently completed, ongoing and planned R&D projects in the topic areas. The abstracts and presentations together with the agendas and the lists of participants are included in the attached CD.

2. CURRENT STATUS OF RESEARCH AND DEVELOPMENT FOR SCWRs

In this section, an overview of R&D status is given, reflecting on related activities that are completed and are ongoing in Member States aimed at establishing the SCWR concepts. It covers R&D areas such as thermal hydraulics phenomena including heat transfer, pressure drop, critical flow, materials and chemistry including materials for fuel cladding and in-core structure and supercritical water radiolysis.

The content of this section is based on the presentations and discussions during the IAEA technical meetings on Heat Transfer, Thermal Hydraulics and System Design for SCWRs, Sheffield, UK, 22–24 August 2016, and on Materials and Chemistry for SCWRs, Řež, Czech Republic, 10–14 October 2016, and is supplemented with relevant information from other related meetings and scientific journals.

2.1. SCWR CONCEPTUAL DESIGNS

As stated in Section 1.1, the SCWR is one of the innovative WCR concepts under development worldwide. It is designed mainly for large scale production of electricity. Operated at supercritical pressure, the SCWRs can achieve much higher core outlet coolant temperature, leading to ~1.3 times higher thermal efficiency compared to the current WCRs. In addition, the system configuration is simplified compared to conventional WCRs. Hence, the SCWR concepts have the potential for improved economics.

The SCWR design can be very flexible: it can be designed as pressure vessel (PV) type reactor like conventional light water reactors (LWRs) or as a pressure tube (PT) type reactor like conventional heavy water reactors (HWRs); the core can be designed as thermal, fast, or even mixed spectrum by adjusting the amount and distribution of a moderator in the core. Various types of SCWR design concepts have been proposed with a combination of reactor type and neutron spectrum. Information on Euratom's High Performance Light Water Reactor (HPLWR), Japan's JSCWR and the Russian Federation's VVER-SCP concepts have been described in an IAEA Technical Document [5]. It has been repeated here for completeness. Updated information on the Canadian and Chinese SCWR designs have been extracted from other publications [6], [7].

2.1.1. Canadian SCWR design

The Canadian SCWR concept is a pressure tube type design [6]. It adopts a direct cycle, which includes a 2540 MW_{th} core that receives feed water at 315 °C and 1176 kg/s and generates supercritical steam at 625 °C at 25 MPa. As illustrated in FIG. 2, the Canadian SCWR core concept consists of a pressurized inlet plenum, a low pressure calandria vessel that contains heavy water moderator and 336 fuel channels that are attached to a common outlet header. A counter-flow fuel channel is adopted to position the inlet and outlet piping above the reactor core so that a complete break of either an inlet pipe or an outlet pipe will not result in an immediate loss of coolant. A non-fuel central flow channel is located at the centre of the fuel channel to increase neutron moderation close to the inner fuel rings.

The coolant flows into the inlet plenum, around the outside of the outlet header (blue arrows in FIG. 2) and then enters the pressure tube extension through a series of slots, into the fuel assembly through a cross over piece (FIG. 2(b)), down through a flow tube in the centre of the fuel assembly, back up through the fuel elements (FIG. 2(c)) and then out through the outlet header.

The fuel channel consists of a pressure tube extending into the moderator, and an extension connecting the pressure tube to the outlet header. All internals of the pressure tube are part of the fuel assembly.

The pressure tube has an open end and a closed end (i.e., a test tube shape). A pressure tube extension is connected to the pressure tube at the top of the tube sheet and incorporates several openings near the interface with the pressure tube to allow coolant entering into the fuel channel and subsequently to the fuel assembly. These openings act as orifices to control the amount of coolant flowing into each channel and to suppress instability. The size of these openings is determined through matching the channel power output to provide an outlet coolant temperature as close to 625 °C as possible.

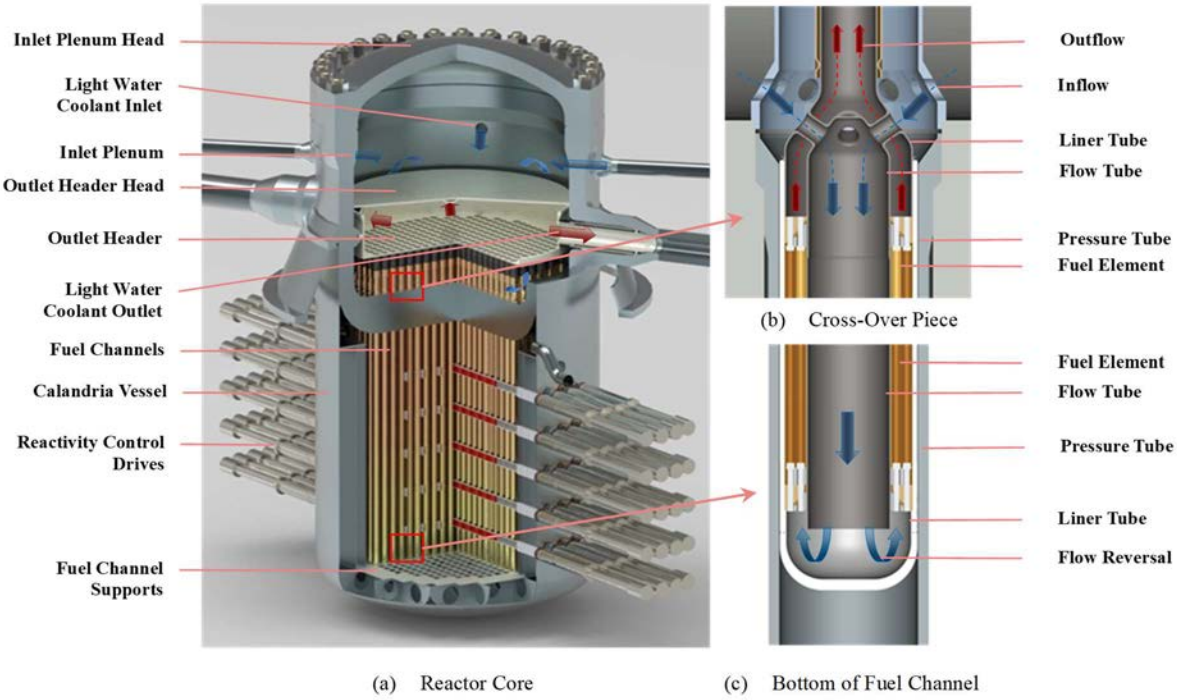


FIG. 2. Canadian SCWR core concept (reproduced courtesy of Leung [6]).

The calandria vessel is a low-pressure vessel that contains the heavy water moderator, fuel channels, reactivity control mechanisms, and emergency shutdown devices. The reactivity control mechanisms located at the sides of the core are shielded, at a minimum, with a similar volume of moderator, with an increasing amount at the reactor centerline due to the curvature of the calandria vessel. The moderator operates at subcooled temperatures using a pumped recirculation system but in case of a station blackout, core decay heat is passively removed using a flashing driven natural circulation loop.

The fuel of the Canadian SCWR is similar to the existing power reactor fuel in that a ceramic pellet produces heat which is transferred through a metallic cladding to the primary coolant. It is a mix of thorium (Th) and plutonium (Pu), which is extracted from the spent fuel of HWRs or LWRs. On average, the weight percentage of Pu in the fuel is 13% [8]. With the high neutron economy of heavy water moderator, other fuel mixes can also be accommodated. Studies have demonstrated the feasibility of using low enriched uranium (LEU) of 7% [9], a mix of LEU with 7.5% Th, a mix of transuranics (TRU) with 21 wt% Th [10], or a mix of 8% Pu, Th and ^{233}U (2 wt%) extracted from the SCWR fuel [11].

The fuel bundle concept consists of 64 fuel elements with 32 fuel elements in each ring (see FIG. 3 for the cross-sectional view). The outer diameter of fuel elements is 9.5 mm in the inner ring and 10 mm in the outer ring. Each fuel element is 6.5 m long housing fuel pellets, an inner filler tube in the plenum area to prevent collapse under external pressure, and a spring to hold the pellets in place but allow for pellet expansion. The active length of a fuel element is 5 m. Each end of fuel element is closed with an end plug, which is welded to the cladding tube. Spacings between fuel elements, between inner ring elements and central flow tube, and between outer ring elements and the inner insulator liner are maintained by wires arranged in a spiral wrap around every fuel element.

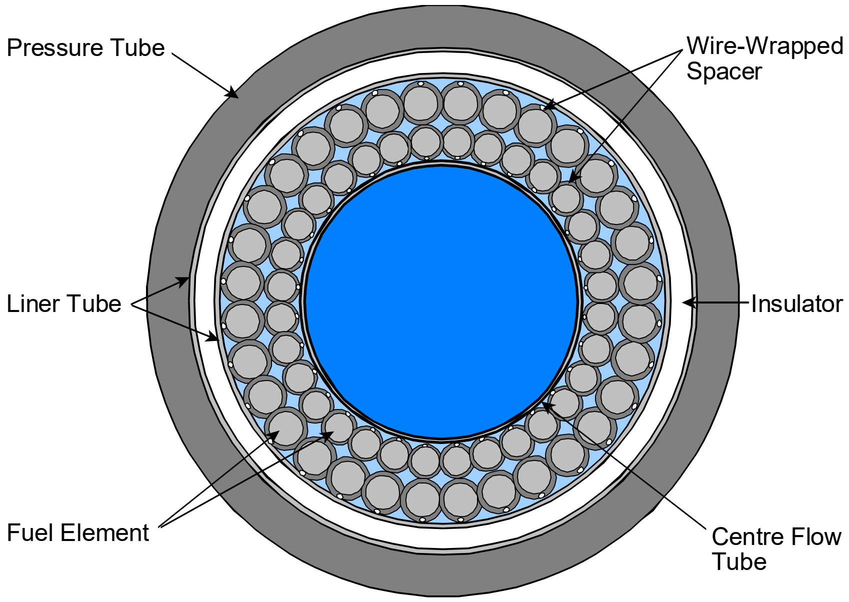


FIG. 3. Cross sectional view of the 64 element fuel bundle concept inside the pressure tube (reproduced courtesy of Leung [6]).

The plant concept for the Canadian SCWR has been evolved from the Advanced Boiling Water Reactors (ABWRs) and the supercritical fossil fired power plant [12]. FIG. 3 illustrates the overall Canadian SCWR plant concept. Additional passive systems have been implemented to improve the safety characteristics of the plant. The inner core structure of the reactor building is the primary containment building, which is a cylindrical steel lined concrete structure. It houses the reactor, high activity components and systems as well as the containment pool. The containment building contains all safety related pressure boundary components. Inlet and outlet pipes penetrating the containment building are equipped with isolation valves so that the radiation release to environment can be isolated and confined inside the containment building. A suppression pool is used to limit the containment pressure. This has led to a reduction in the volume of the containment building compared to current fleet of nuclear reactors.

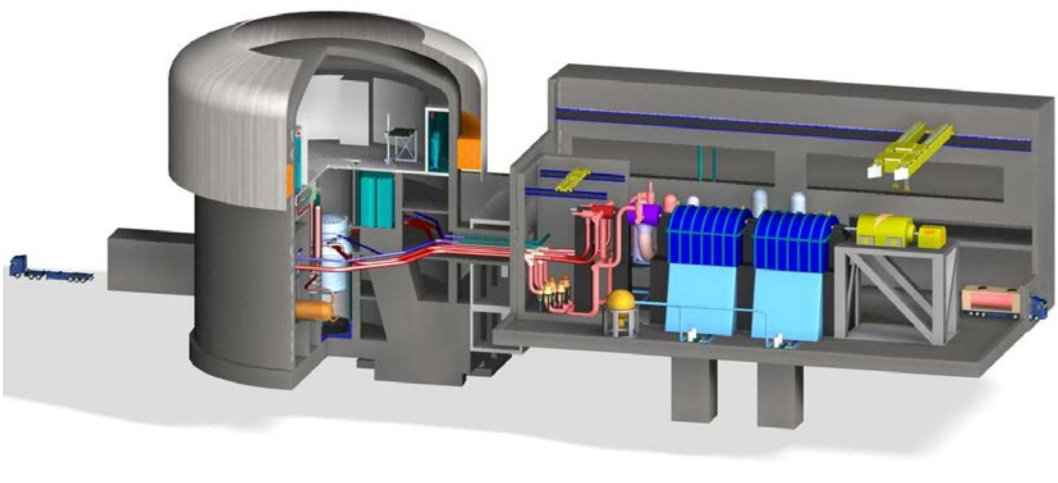
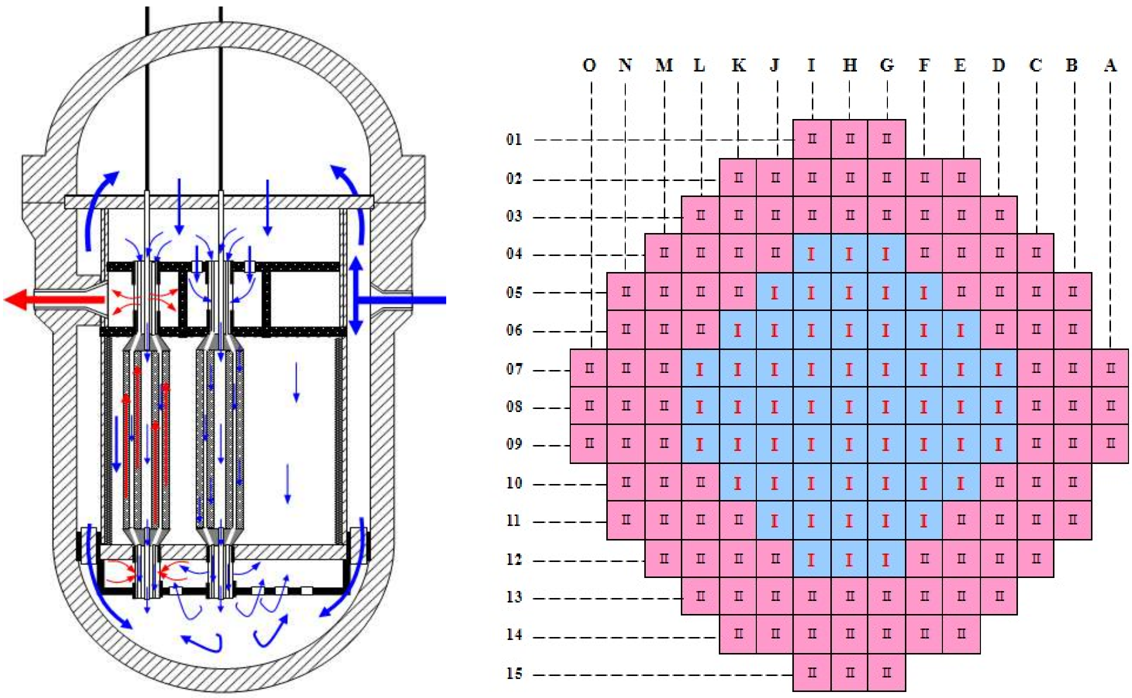


FIG. 4. Plant concept for the Canadian SCWR (reproduced courtesy of Yetisir [12]).

2.1.2. Chinese SCWR (CSR1000) design

The Chinese SCWR concept, CSR1000, has been developed by the Nuclear Power Institute of China (NPIC) [7]. The primary circuit of CSR1000 is a direct cycle system consisting of a two-pass, thermal neutron reactor cooled and moderated by light water, two primary loops connected with supercritical turbine and feedwater pumps etc. The primary circuit is also interfaced with passive safety features. The core thermal power is 2300 MWt with system thermal efficiency of 43.5%, leading to system output electrical power around 1000 MWe. The primary circuit operates at 25.0 MPa. The feedwater temperature is 280 °C, and the average core outlet coolant temperature is about 500 °C. The CSR1000 reactor core consists of 157 fuel assemblies. The core coolant flow rate, 1189 kg/s, is significantly lower than those of current LWRs since the enthalpy rise in the core is much higher than that of traditional LWRs.

The two-pass core design of CSR1000 is shown in FIG. 5. Fifty seven fuel assemblies are located in the core center, signed as ‘I’ style fuel assembly; the rest lie in core periphery, marked as ‘II’ style fuel assembly. In order to meet refueling requirement, the structure of ‘I’ style fuel assemblies must be the same as that of ‘II’ style fuel assemblies. Considering two-pass core arrangement, moderator water passage and coolant passage must be separated from each other. In addition, moderator water and coolant should have enough flow area, and there is no interference with structure among the different components of fuel assembly.



(a) Reactor coolant flow inside reactor Pressure vessel (RPV).

(b) Cross sectional view of the CSR1000 core.

FIG. 5. Two-pass core design of CSR1000 (reproduced courtesy of Huang [7]).

The CSR1000 containment is pressure suppression type as shown in FIG. 6. The containment is divided into areas: reactor pit, wet well, dry well. The reactor pit contains a reactor vessel. The dry well encloses the reactor coolant system (RCS) components and some safety system components. The wet well has a large condensation pool for suppressing the containment when the steam is discharged from the RCS to the containment.

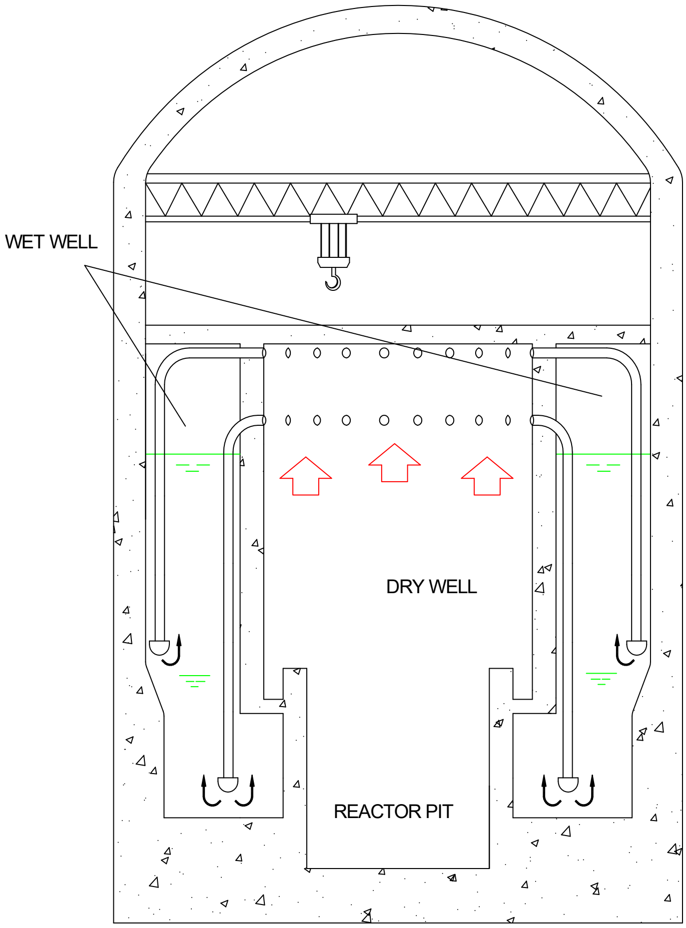


FIG. 6. CSR1000 pressure suppression type containment (reproduced courtesy of Huang [7]).

There are a lot of pressure balance pipes set in the containment. When the dry well is pressurized by the steam which is discharged from the main system, the steam will be discharged to the condensation pool and condensed via the balance pipes. In this way, the containment can be suppressed after the automatic depressurization systems (ADS) operation or a LOCA.

2.1.3. European SCWR (HPLWR) design

The High Performance Light Water Reactor (HPLWR) concept has been developed by a consortium of 12 organizations from eight (8) European countries. 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%. The HPLWR is designed as a thermal neutron spectrum reactor using light water as a moderator and reactor coolant. The plant consists of a pressure vessel type, once through reactor and a direct Rankine cycle system.

The RPV contains core, mixing plenums, and control rods [13]. 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. 7.

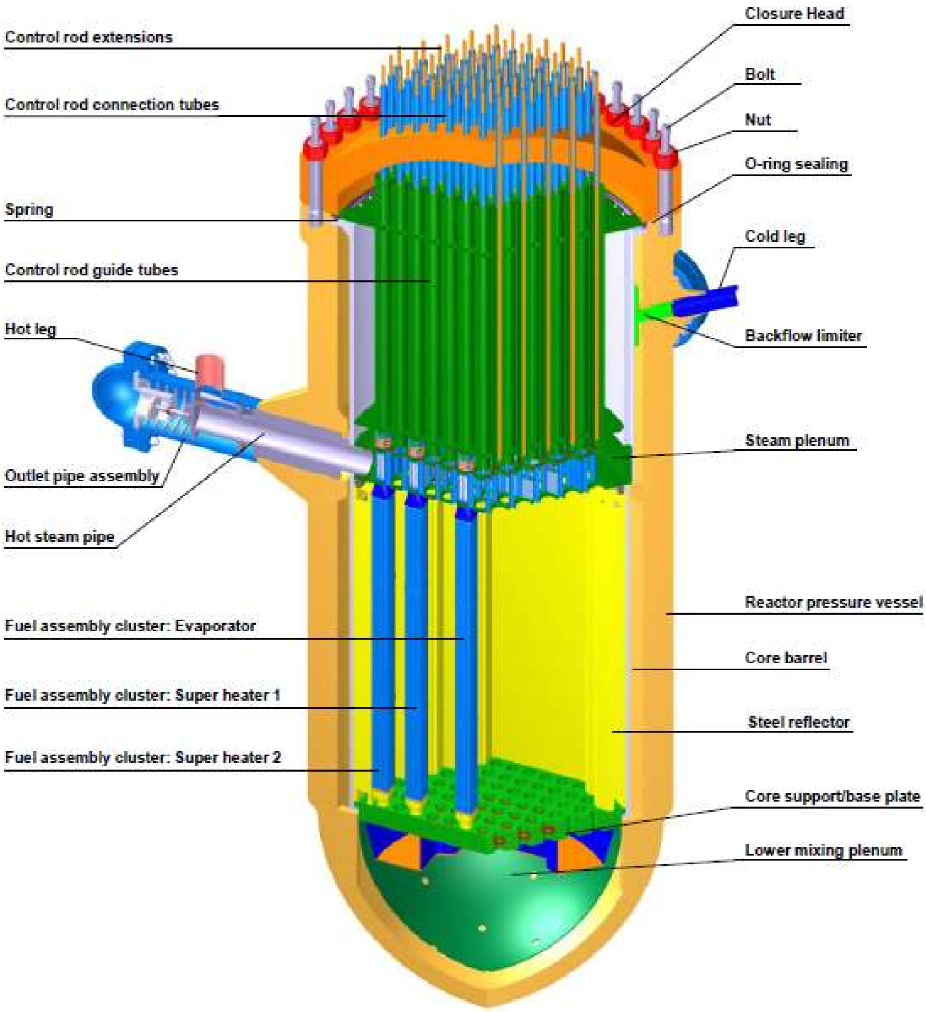


FIG. 7. HPLWR RPV and core internals (reproduced courtesy of Schulenberg [13]).

The HPLWR core design concept assumes that 50% of the coolant supplied through four (4) inlet flanges to the RPV is taken first as moderator water to run upwards to the closure head, then downwards through control rod guide tubes and through the central water boxes inside the housed assemblies, to be released through the foot pieces of the assembly clusters to the gap volume between the assembly boxes. From there, it rises upwards to serve again as moderator water outside the assembly boxes. It is collected at the top of the core to cool the radial core reflector with a downward flow, before it is mixed with the remaining 50% of the coolant in the lower mixing plenum underneath the core. The following three heat up steps comprise an evaporator region formed by 52 assembly clusters in the core centre, where the coolant changes its density from liquid like to steam like conditions, followed by an upper mixing chamber above the core. Another 52 assembly clusters with downward flow surround the evaporator region and serve as the first super heater. After a second mixing in an annular mixing chamber underneath the core, the coolant is finally heated up to 500 °C in a second super heater region formed by 52 assembly clusters at the core periphery. The core arrangement is shown in FIG. 8.

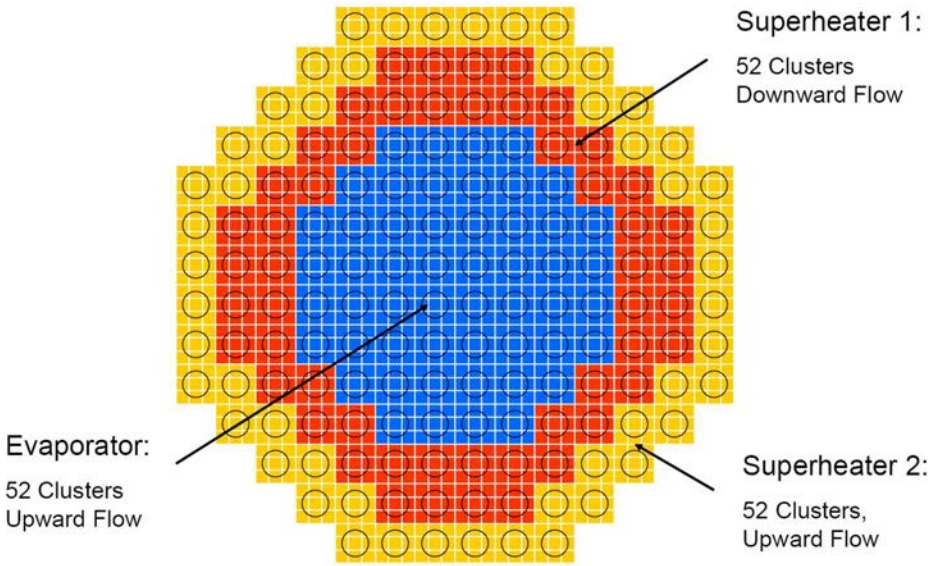


FIG. 8. Arrangement of the evaporator and super heater regions in the HPLWR core (reproduced courtesy of Schulenberg [13]).

A design proposal for the HPLWR containment is shown in FIG. 9. It is made from reinforced concrete, equipped with an inner steel liner and a pressure suppression system. The design pressure of the containment is considered to be in the range of about 0.3 to 0.4 MPa. Containment isolation valves for each of the four 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 scrammed and the depressurization valves release steam through eight spargers into four upper pools, removing the residual heat until at least one of the four redundant, active low pressure coolant injection pumps in the basement of the containment becomes available. 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 into which sixteen open pressure suppression tubes are running. As an additional passive high pressure safety system, it is proposed 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 to the spargers is starting the steam injectors within the first 10 seconds. This design proposal, however, is still to be verified. As a backup alternative to cool down the core at high pressure 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 (RHR) from the containment can also be provided by containment condensers to the spent fuel pool above the containment.

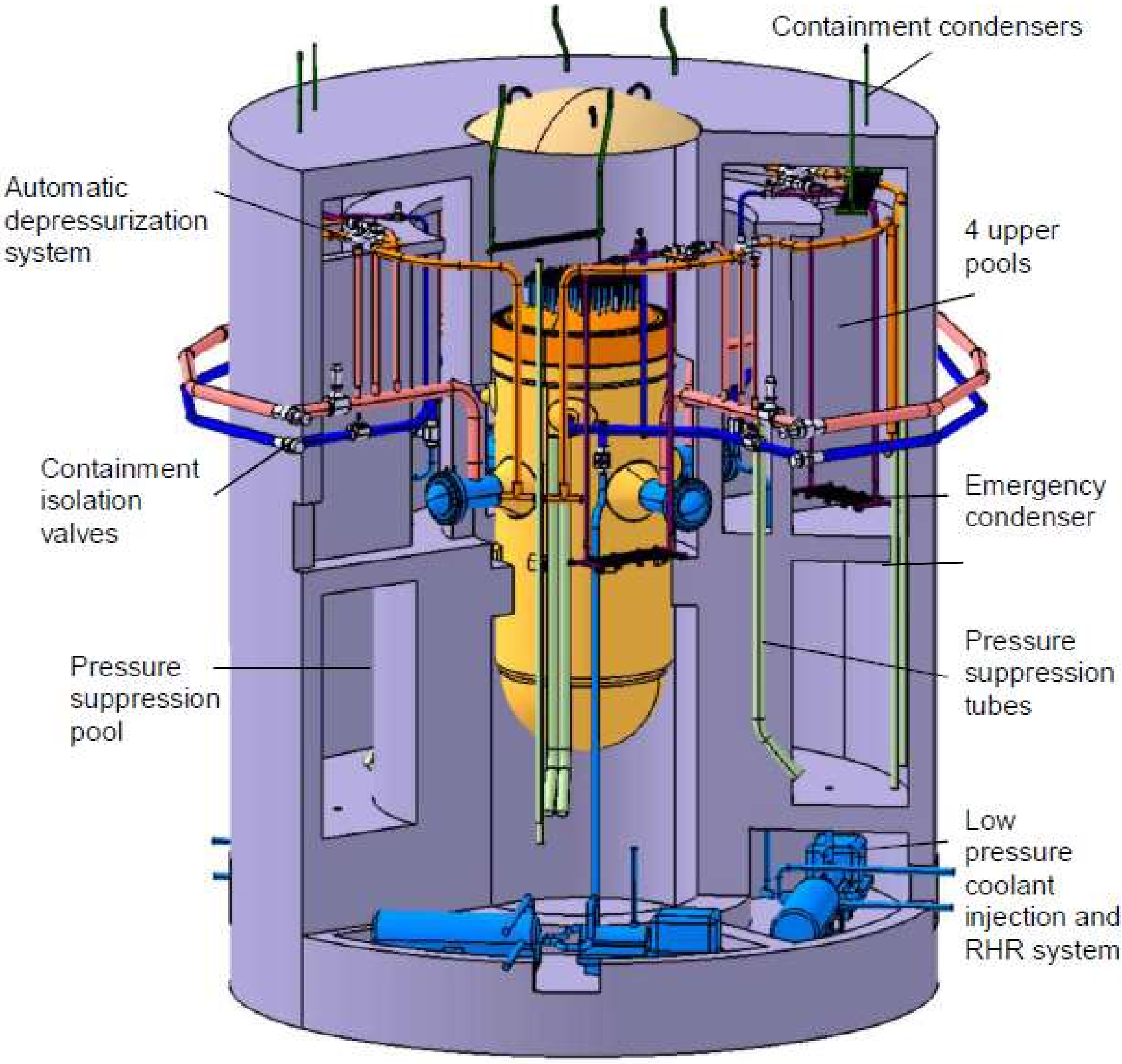
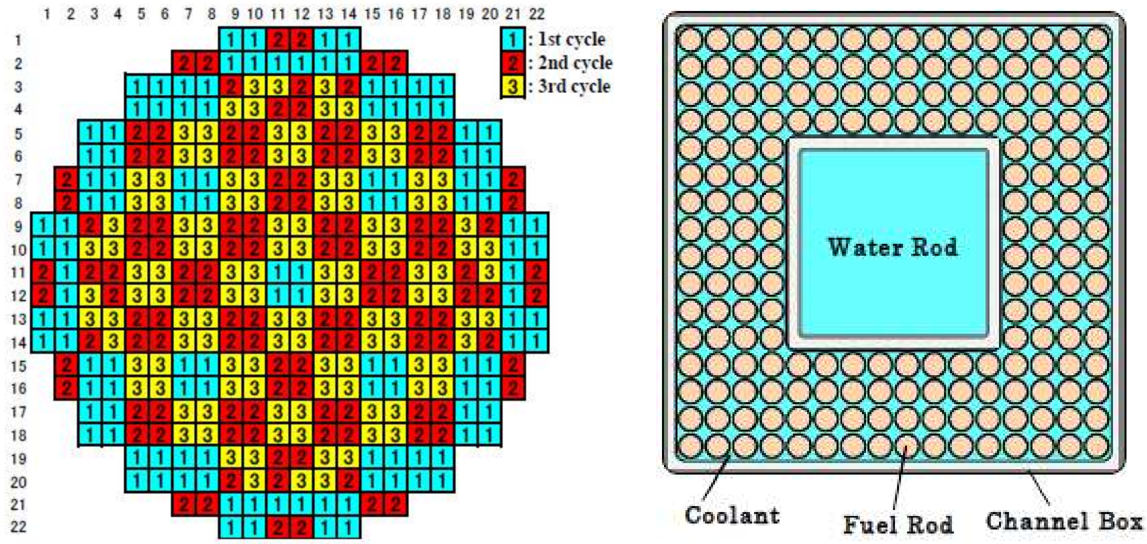


FIG. 9. HPLWR reactor building configuration (reproduced courtesy of Schulenberg [13]).

2.1.4. Japanese SCWRs design

The JSCWR concept has been developed by a Japanese consortium consisting of eight (8) organizations under the financial support of the government of Japan. The basic philosophy of the JSCWR development is to utilize proven light water reactor and supercritical fossil fuel power plant technologies as much as possible to minimize the R&D cost, time and risks. The JSCWR is designed as a thermal neutron spectrum reactor using light water as a moderator and reactor coolant [14]. The JSCWR plant consists of a pressure vessel type, once through reactor and a direct Rankine cycle system. The reactor has no recirculation in the vessel, and reactor coolant is directly delivered to the turbine system.

The reactor core is operated at 25.0 MPa. The feedwater temperature is 290 °C, and the average core outlet coolant temperature is 510 °C. The coolant flow rate, 2105 kg/s, is significantly low since the enthalpy rise in the core is high compared to those of current WCRs. The reactor core is cylindrical in shape consisting of 372 fuel assemblies. Each fuel assemble stays in the core for three cycles. The cross-sectional view of the core is shown in FIG. 10(a) together with its loading pattern. A fuel assembly consists of 192 fuel rods and a square water (moderator) rod in the center, surrounded by a square channel box (137 × 137 mm) as shown in FIG. 10(b). The fuel rods contain UO_2 pellets like LWR fuels in the modified stainless steel cladding. In the water rod, low temperature water flows downward to keep enough moderation in the core. The active fuel length is 4.2 m, which is a little longer than typical LWR fuels, to reduce the linear heat generation rate. The total fuel length is about 5.8 m. Control rods are used for primary reactivity control. The control rod drives are mounted on the bottom of the RPV. Cruciform control rods are vertically inserted and withdrawn from the core by the control rod drives. To ensure adequate shut down margin and to minimize the local peaking during the entire operation cycle, gadolinia (burnable poison) is incorporated in the fuel. The ^{235}U enrichment for the equilibrium core exceeds 7% to achieve similar discharge burnup as with current LWR fuels. This high enrichment is mainly due to relatively high neutron captures of the structural materials especially fuel claddings and channel boxes. The structure of the JSCWR RPV is similar to that of PWR. The inner diameter is about 4.8 m; the total inside height is 16.5 m.



(a) JSCWR core and its loading pattern. (b) Cross sectional view of fuel assembly.

FIG. 10. Cross-sectional view of the JSCWR core and its loading pattern (reproduced courtesy of Yamada [14]).

The JSCWR safety philosophy is based on that of the advanced LWRs, which reflects experiences and lessons learned of the past and current LWRs. The design philosophy for safety and reliability are as follows:

— Maximum utilization of the matured, proven technologies that have been accumulated in the successful commercial operation of LWRs as well as supercritical pressure fossil fuel power plants;
— Safety system development based on inherent feature of water cooled reactor and well developed LWR safety technologies. The inherent feature includes negative void (density) and Doppler coefficients. The well developed LWR safety technologies mainly include reactivity control systems and emergency core cooling systems (ECCS).

Safety systems mainly consist of high pressure auxiliary feedwater systems (AFS), Automatic Depressurization System (ADS), and low pressure coolant injection system (LPCIs) that also work as residual heat removal system (RHR). Reactor scram, AFS and LPCI are actuated by low core flow rate signals instead of low water level signals, which is commonly used in current BWRs [14].

The turbine system of the JSCWR consists of one dual exhaust high pressure (HP) section, one dual exhaust intermediate pressure (IP) section and three dual exhaust low pressure (LP) sections (that is, a six flow, tandem compound system). The cycle uses a moisture separator to reduce the wetness fraction of steam to prevent damages of the LP turbine blades. The turbine system could be simplified by adopting a combined high pressure and intermediate pressure turbine, which is under development in the fossil fuel power industry, if the steam flow rate is low enough. The adoption of the combined turbine as well as full speed turbine system might reduce the length of total turbine system and hence reduces the volume of turbine building. FIG. 11 shows the balance of plant (BOP) system of the JSCWR and its heat balance.

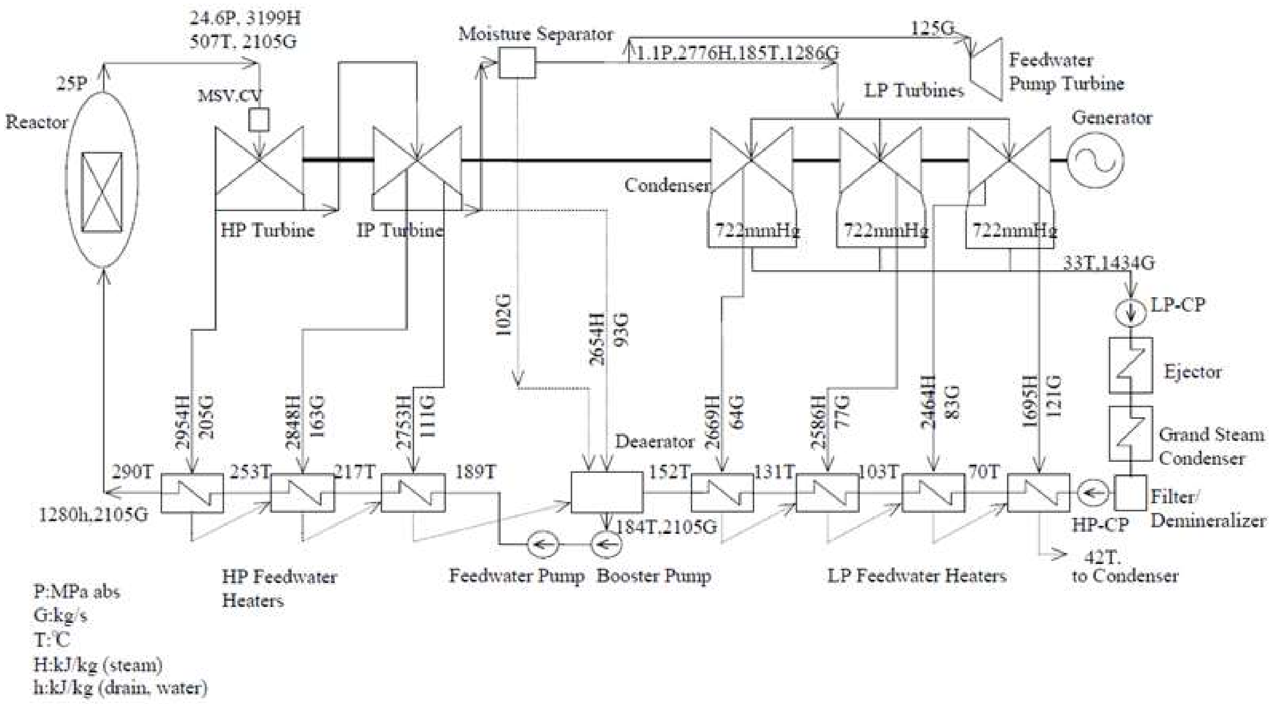


FIG. 11. JSCWR balance of plant system (reproduced courtesy of Yamada [14]).

2.1.5. Russian SCWRs design

The VVER-SCP[1] concept is developed by OKB GIDROPRESS, Russian Federation. Its reactor vessel and internals are shown in FIG. 12 [15]. The nominal electric power of 1700 MW is considered. Fuel conversion factor is targeted to be close to one (0.9–1.0) when fast resonance neutron spectrum is realized.

Two versions of the reactor that differ in core coolant flow diagram: single pass flow diagram (FIG. ***13***(a)) and double pass flow diagram (FIG. 13(b)). The usage of the single pass flow diagram with once through upward coolant flow in the reactor core results in a design simplification and improves the plant safety under conditions with natural coolant circulation. In the double pass flow diagram the coolant moves downwards along the peripheral circular part and moves upwards in the central part. The application of the double pass flow diagram ensures negative void reactivity effect within the entire fuel cycle without special technical solutions; improvement of the conditions of fuel rods cooling due to an

[1] SCP: supercritical pressure

increase in coolant flow velocity; reduction in temperature differentials along the core height; shift of the point with coolant pseudo critical temperature into the core lower part, in which relatively small heat fluxes take place (deposits of impurities on fuel rods claddings and the conditions of deteriorated heat transfer are expected in the pseudo critical temperature field); coolant mixing in the lower chamber, which decreases the unevenness of coolant heating at the outlet of the fuel assembly central part.

The core of both reactor concepts consists of jacketed hexahedral fuel assemblies. The fuel assembly consists of fuel rods, a jacket tube, guide tubes for the rod cluster control assemblies, absorber rods to move inside them, elements to create neutron spectrum using zirconium hydride, spacer elements and structural components (as illustrated in FIG. 14). One of the specific features is a tight fuel grid: the distance between the fuel rods is 1.3 mm.

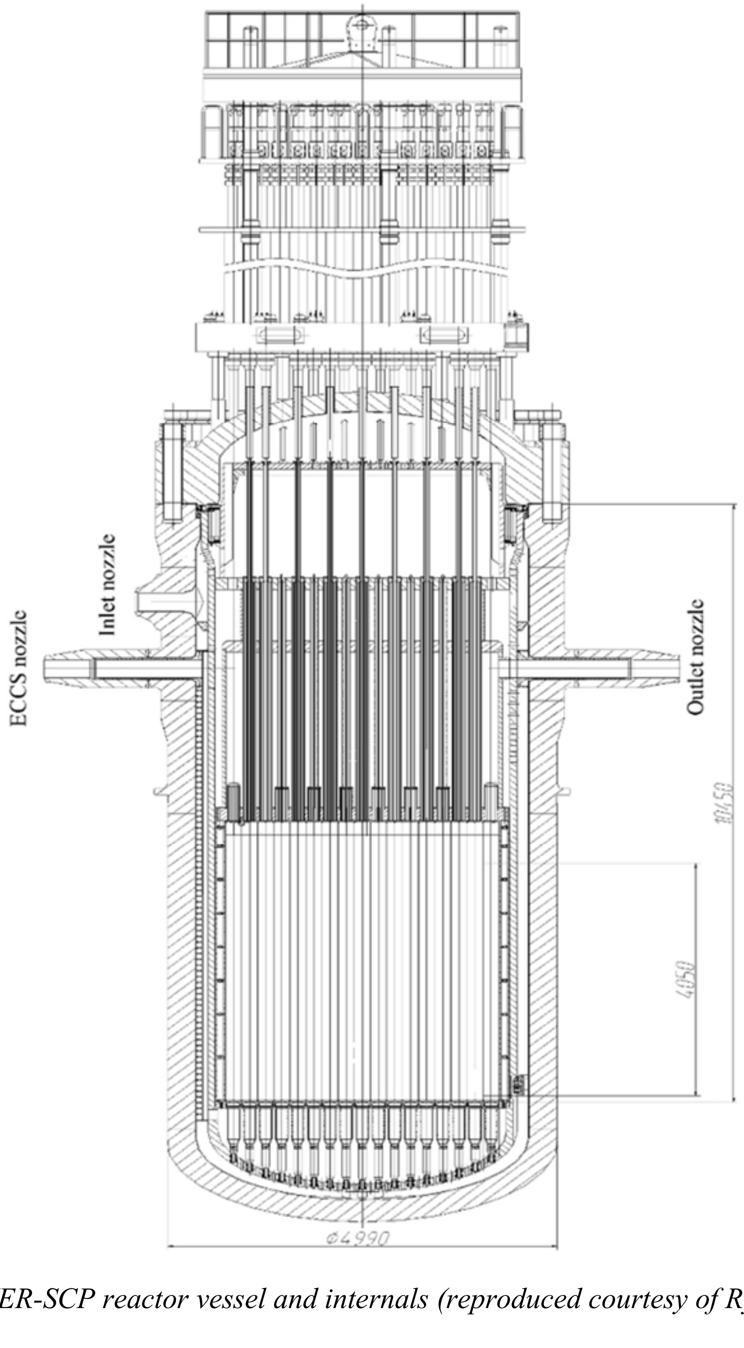
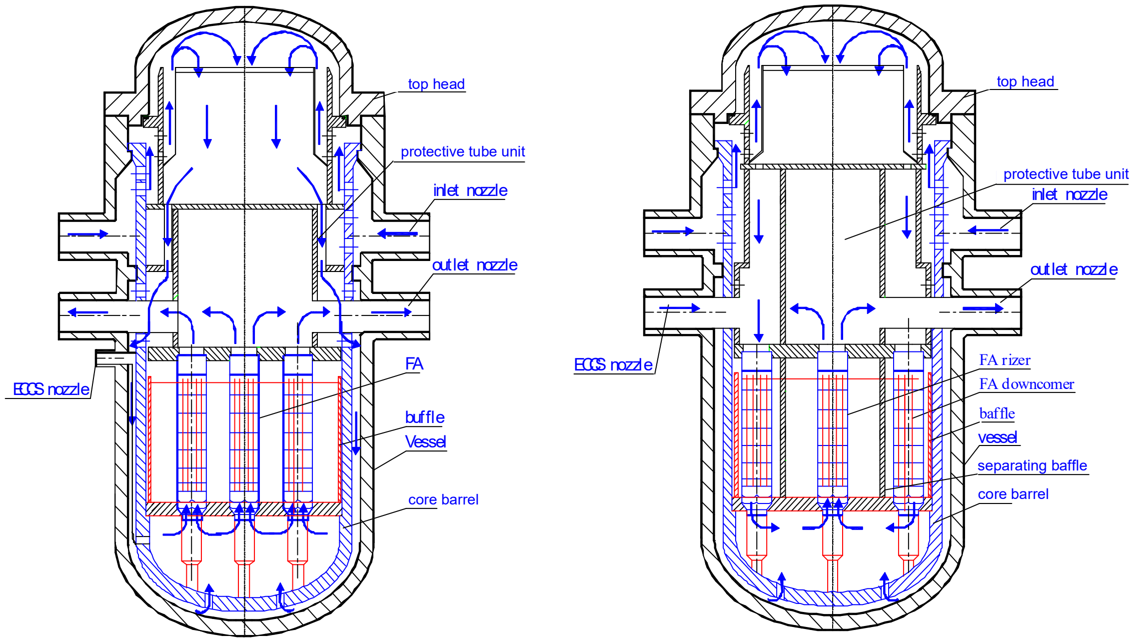


FIG. 12. VVER-SCP reactor vessel and internals (reproduced courtesy of Ryzhov [15]).



(a) Single pass flow diagram. *(b) Double pass flow diagram.*

FIG. 13. Direction of coolant flow in VVER-SCP core (reproduced courtesy of Ryzhov [15]).

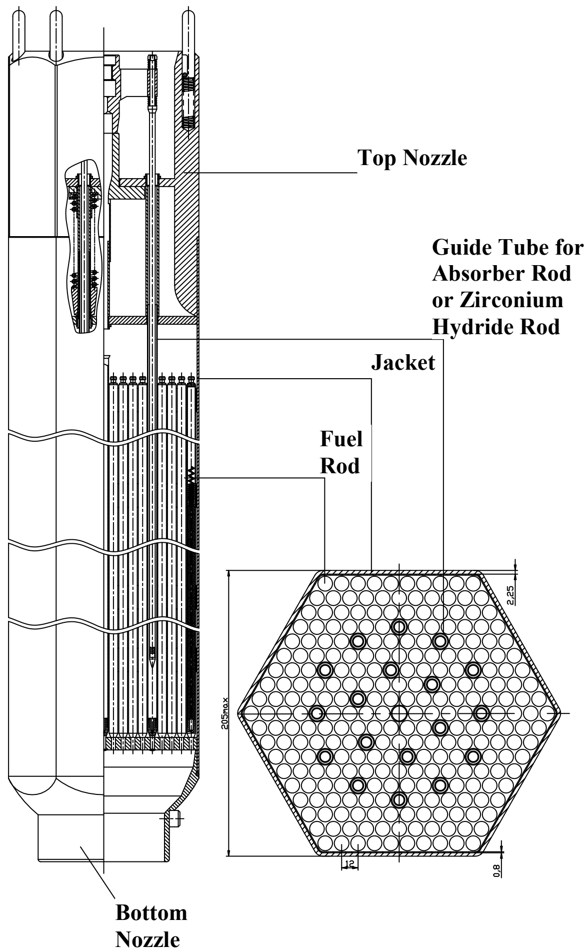


FIG. 14. Jacketed fuel assembly of the VVER-SCP (reproduced courtesy of Ryzhov [15]).

2.1.6. Comparison of SCWR designs

Table 1 compares the features of various SCWR design concepts that are being developed worldwide and described above. A Canadian SCWR is a pressure tube type (PT), while the others are pressure vessel type (PV). Regarding the neutron spectrum, some design concepts have thermal spectrum, and others have fast spectrum. The operation pressure, the core outlet coolant temperature and the thermal efficiency have been discussed previously. Most of the design concepts adopt double or triple core flow pass which means that reactor coolant passes through the core region in between the core inlet and core outlet twice or three times, unlike single pass means that reactor coolant passes through the core region only once between the core inlet and the outlet like in conventional PWRs.

TABLE 1. COMPARISON OF VARIOUS SCWR DESIGN CONCEPTS

	CANADIAN SCWR	CSR1000	HPLWR	JSCWR	VVER-SCP
Reactor type	PT	PV	PV	PV	PV
Neutron spectrum	Thermal	Thermal	Thermal	Thermal	Fast
Coolant	Light water	Light water	Light water	Light water	Light water
Moderator	Heavy water	Light water	Light water	Light water	None
Normal operation pressure (MPa)	25	25	25	25	24.5
Core outlet coolant temperature (°C)	625	500	500	510	540
Plant thermal efficiency (%)	48	43.5	43.5	42.7	43–45
Core coolant	Light water	Light water	Light water	Light water	Light water
Core flow pass	Single	Double	Triple	Single	Double (Single)

2.1.7. Development stage of SCWR design concepts

Up to now, the Canadian SCWR concept, the European SCWR concept (HPLWR) and Japanese SCWR concept (JSCWR) have been reviewed by international expert groups from the GIF SCWR System and assessed technically feasible. The CSR1000 core concept has been established. NPIC is completing the plant concept and has planned to host an international peer review of the CSR1000 concept. It has also proposed to design a small demonstration SCWR of 150 MWe (CSR150).

The design of an in pile supercritical pressure water loop for material samples irradiation (excluding fuels) has been completed and the licensing required documentation submitted to a local nuclear safety authority. The goal of the facility is to provide the opportunity to contribute to the design and licensing of a supercritical pressure facility, which is a prerequisite for the prototype reactor design.

2.2. THERMAL HYDRAULICS

Mainly because of rapid changes in thermo-physical properties across the pseudocritical temperature range and no strict distinction between liquid and gas phases, thermal hydraulics behavior of supercritical pressure fluids can be quite different from those at subcritical pressures. For example, heat transfer enhancement at near pseudocritical temperatures and heat transfer deterioration at high heat flux and low mass flux conditions are observed only in supercritical pressure fluids.

The thermal hydraulics R&D of supercritical pressure fluids is one of the most important areas to establish and analyze the SCWR concepts. Extensive efforts have been devoted to the subject matter internationally for the past 15 years. The main interest so far has been focusing on fundamental research to understand thermal hydraulics phenomena of supercritical fluids (e.g., heat transfer tests with tubes or annuli). In addition, prediction methods have been developed supporting the SCWR concept development.

The thermal hydraulics R&D in support of SCWR development focuses mainly on heat transfer, pressure drop (or hydraulics resistance), stability and critical flow [16]. Information on heat transfer and pressure drop is required for analyses of normal operating conditions and postulated accident scenarios. Heat transfer calculations are applied to establish the fuel and cladding temperatures. Pressure drop analyses are needed to establish the mass flowrate in the core and the pump capacity/requirement. In addition, these analyses help to establish the subchannel flow in fuel assembly. Stability calculations are essential to identify the operating and startup conditions for SCWRs in view of the significant variation in density in the core, which could lead to dynamics instability. Another interesting SCWR characteristic is the strong coupling between thermal hydraulics and neutronics. Instability could lead to power fluctuations within the core. The ability to depressurize rapidly is a requirement for SCWR in mitigating any upset conditions. A pressure relieve valve is one of the major components in the ADS. Design of this valve requires the critical flow behavior at supercritical pressures. In addition, information on this behavior is also needed for the analyses of postulated large break loss of coolant accident (LOCA).

2.2.1. Heat transfer to supercritical pressure fluids including the subchannel mixing effects

The fuel design criterion for the SCWR is based on the cladding temperature limit because phase change is not present in the core during normal operation. The lack of qualified experimental data on heat transfer for supercritical water flow in rod bundles has been identified as a significant challenge to the SCWR design. This is due to the possibility of drastic deterioration of heat transfer in the vicinity of critical point. Fundamental understanding of thermal hydraulics characteristics has relied on experimental information obtained with tubes, annuli, and bundle subassemblies.

Information about supercritical pressure heat transfer characteristics of water is of prime interest. Several heat transfer databases for supercritical water flow have been assembled. Most supercritical heat transfer correlations were developed using subsets of these databases. Improved correlations have been derived using combined database to provide better prediction accuracy and more appropriate parametric trends than the existing correlations.

Supercritical heat transfer data for surrogate fluids have been applied as supplemental information to the water data. Testing with surrogate fluids is usually more flexible due to relatively less severe conditions compared to water. Most studies employed carbon dioxide and Freon as surrogate fluids to examine separate effects on heat transfer characteristics. Direct applications of surrogate fluid heat transfer data in SCWR design and safety analyses require the development of fluid to fluid modelling parameters to relate the surrogate fluid heat transfer coefficients to water equivalent values.

Well established and validated correlations for heat transfer coefficient (HTC) have to be included into analytical models, for example, for the performance evaluation of SCWRs and accident analyses for licensing. Experimental studies have shown that there is a heat transfer enhancement (I) for supercritical fluid near pseudocritical temperature at relatively low heat flux to mass flux ratios. The HTC peak decreases as this ratio increases. At very high values of heat flux, a peak in wall temperature appears due to heat transfer deterioration (HTD). Many universities and research institutes focused on advancing the understanding of thermal hydraulics phenomena and developing semi-analytical theories and correlations. Several investigators have carried out the experiments using water and surrogate fluids (such as CO_2 and R-134a) due to similarity in their thermo-physical properties [17], [18], [19]. It was observed that the bulk fluid enthalpy at which peak wall temperature appears is different for different test values of heat flux, mass flux and inlet temperature. Effects of inlet temperature are found to be significant for HTD but details on this effect are not found in literature. Experimental wall temperatures, available to be found in open literature, were measured by thermocouples positioned at regular intervals. However, since the HTD is shown to result in very steep temperature changes, highly local temperature measurements are desirable. The HTC correlations are available to prediIHTE satisfactorily. However, HTD predictions from available correlations are unsatisfactory and therefore, better correlations are required to predict HTD.

The accuracy of HTC is very important in estimating correctly a peak cladding temperature (PCT). The uncertainties in predicting heat transfer coefficients, however, are still large, and the efforts to reduce them continue. Very recently, it is reported that the heat transfer correlation proposed by Chen and Fang show a good agreement with experimental data obtained with supercritical water in tubes and rod bundles [20].

On the other hand, it is pointed out from the viewpoint of the design of SCWR cores that an accurate prediction of the PCT (which is the limiting parameter) is of the most interest compared to HTC (which is a means to calculate the PCT). In particular at the vicinity of the pseudocritical temperature, where any significant uncertainty in HTC would lead to only a minor impact in PCT due to the small difference between cladding and bulk fluid temperatures. This implies that the focus should be on improving the prediction accuracy of PCT in bundles. An essential component for improvement is the compilation of relevant data from bundle subassembly experiments. Such experiments will be costly and time consuming. Significant support and involvement of the industry and/or governments will be needed.

Subchannel and system analysis codes are being developed as design and analysis tools for SCWR applications just like those for conventional WCRs [21], [22], [23]. These analytical tools are expected to play a major role in the development of SCWRs. Water properties at supercritical pressures and relevant closure relationships have been implemented into these codes to support design and optimization of the SCWR cores and fuel assemblies.

In addition, the conventional computer fluid dynamics (CFD) codes and direct numerical simulation (DNS) are used to capture the phenomena that are difficult to clarify only through the experiments. It is recognized that it will be very difficult to replace subchannel or system analysis codes in near future with conventional CFD codes. However, it is expected that the CFD codes will be used to support design in the future. The DNS may not be used to directly support design in near future, but it is expected to provide very useful information such as subchannel mixing and heat transfer mechanism near heated walls, which will support design and analysis indirectly. Both CFD and DNS are expected to reduce over conservatisms in the designs.

2.2.2. Pressure drop

Pressure losses in heat transport system have a direct impact on pump capacity and power output, affecting capital and operating costs. In addition, these losses affect local flow conditions along the fuel assembly impacting heat transfer characteristics. Depending on the design, the pressure loss consists of frictional loss, form losses (such as valves and elbows), head loss due to gravity, and loss due to acceleration. A significant portion of the overall pressure loss occurs in the core region (particularly over the bundle assembly). Therefore, it is important to evaluate the core pressure loss accurately. Outside of the core region, the bulk fluid temperature does not vary considerably over various sections.

The hydraulics resistance is generally relatively low in SCWR due to low mass flow rate. The highest hydraulics resistance is associated with the fuel assembly, which is more complex than that in current reactors. The pressure drop over a bundle assembly consists of several components: skin friction, spacing devices, acceleration, and gravity. In general, the pressure drop due to skin friction is the largest component. The pressure drop due to acceleration can be significant due to the large change in density, particularly over the pseudocritical point. Therefore, it must be considered separately at conditions of interest. With the fuel heating, the enthalpy and coolant temperature increase leading to changes in fluid properties. In addition, the sharp change in fluid properties at the pseudocritical point makes it inapplicable to use the average fluid properties over the fuel assembly in calculation.

Experimental data on pressure drop have been obtained as well from the heat transfer tests, using the tubes, annuli and bundle subassemblies in water or surrogate fluids [24], [25]. These overall pressure drop measurements covered both the liquid like and steam like regions and often included other separate effects (such as sudden contraction, sudden expansion and fittings) making it challenging to analyze the data, and leading to high uncertainty.

The pressure loss calculation over the fuel assembly is separated into two regions based on pseudocritical enthalpy (or temperature). This is an analogy to current reactor safety analyses with the departure from nucleate boiling or dryout as the transition point. Separating the fuel assembly into two regions would minimize the effect of rapid fluid property change at pseudocritical point.

Pressure drop relationships were compiled and described in an IAEA publications for advanced WCRs [5], [26], [27]. These relationships cover both the single phase and two phase flow. The single phase pressure drop models are likely applicable to supercritical flow, while the two phase pressure drop models are relevant for startup, shutdown, and pressure transient analyses.

2.2.3. Critical flow

When a fluid discharges from a high pressure and high temperature system, a ‘choking’ or critical condition occurs, and the flow rate no longer increases as the downstream pressure is further decreased. During a postulated LOCA in PWR, the break flow will be subject to such condition. An accurate estimate of critical flow rate is important for the evaluation of reactor safety, because this flow rate controls the loss of coolant inventory and energy from the system, and thus has a significant effect on the accident consequence.

A depressurization event or LOCA is particularly of a concern for the SCWR safety due to the lower coolant inventory compared to a typical PWR for the same power output. This lower coolant inventory would result in a faster transient response of the reactor dominated by the discharge flow rate. Critical flow is coupled with drastic depressurization and vaporization, and the flow rate is heavily dominated by the vapor content or quality of the vapor, which is closely related with the onset of vaporization and the interfacial interaction between phases. This presents a major challenge to estimate the flow rate due to the lack of the knowledge about these processes, especially under the conditions specific to SCWR.

High pressure depressurization leading to critical flow has been investigated extensively, in both experimental and theoretical ways [28], [29]. These studies have mostly been conducted with subcritical pressure or ideal gases well above the vapor dome, but little work has been done near or above the thermodynamic critical point. The experiments have also been performed with other supercritical fluids such as CO_2, which can be considered a suitable surrogate fluid to water. These surrogate fluids offer a close approximation to the physics that occur during the SCW depressurization thus the study can be used as a qualitative indication of water flow and the test tool of physical models.

Based on limited data of supercritical fluids, the critical flows at conditions above pseudocritical point seem to be fairly stable and consistent with the subcritical homogeneous equilibrium model (HEM) predictions. Further, in this regime the flows tend to be lower than those in the two phase region. Thus, the major difficulty in predicting depressurization flow rate remains in the region where two phases coexist i.e., within the two phase portion of the vapor dome. In this region, the flow rate is strongly affected by the nozzle geometry and tends to be unstable. Various models for this region have been developed with different assumptions, e.g. the HEM and Moody mode, and the Henry–Fauske non-equilibrium model, and are currently used in subcritical pressure reactor safety design [28]. It appears that some of these models could be reasonably extended to above the thermodynamic pseudocritical point. The more stable and smaller flow observed in conditions above the pseudocritical point suggests that even though SCWR designs have smaller coolant inventory, the safety implications of a LOCA and the subsequent depressurization from supercritical conditions may not be as severe as expected and the limiting accident scenario may be driven by critical flow within the subcritical pressure region.

2.2.4. Flow instability

The design of SCWRs poses new challenges in relation to the prediction of some basic phenomena, whose characteristics are relatively well known at subcritical pressure.

In particular, stability and natural circulation phenomena observed in single phase conditions at subcritical pressures become more complex at supercritical pressures because of the strong variability of thermodynamic and thermo-physical properties. Moreover, categories adopted for dealing with two phase flow phenomena must be updated to address the supercritical pressure flow regimes, having in principle a simpler structure than the corresponding two phase flow ones, though they anyway involve uneven distributions of lighter and heavier fluids within the channel cross section.

Flow instabilities and natural circulation at subcritical pressures have been the subject of major experimental and modelling efforts. Some of these studies belong to both aspects, since a quite populated intersection of the two categories is represented by the interesting domain of natural circulation instabilities [5].

In single phase flow, in particular, apart from turbulence and acoustic effects, the instabilities mainly occur in natural circulation systems, being driven by a mechanism explained with effective evidence based arguments. It is suggested that pockets of fluid with perturbed temperature may emerge from the heat source and/or the sink, affecting the fluid residence time along the loop and being damped or amplified as a consequence of the subsequent passages through the heater and the cooler [5]. This mechanism was proposed on the basis of numerical computations and was confirmed later on by further experimental and computational studies, focusing on different interesting aspects affecting the phenomenon. A remarkable feature of the observed behaviour is its chaotic character, mainly caused by the fact that the growth of flow perturbations frequently leads to flow reversal; as a consequence, the system is seen to continuously switch from clockwise to counter clockwise flow directions and vice versa, showing a high sensitivity to initial conditions.

Two phase flow instabilities are mainly connected to the behaviour of boiling channels that are recognized to be susceptible to different instability mechanisms, having a ‘static’ or a ‘dynamic’ nature. The discrimination among the two categories of instabilities is introduced considering whether their occurrence can be explained on the basis of the static internal pressure drop vs. flow characteristic of the system. The Ledinegg ‘excursive’ instability mechanism is representative of the category of static instabilities, while density wave, pressure drop oscillations and thermal oscillations are mechanisms which belong to the second category. Of these instabilities, density waves represent the most relevant instability mechanism for nuclear reactor applications since, in combination with neutronics feedback, it is responsible for observed space and time oscillations of boiling water reactor (BWR) core power. These instabilities have been the subject of extensive studies, summarised in review papers and state of the art reports. Basic experimental investigations allowed grasping the main features of involved phenomena, gaining the basic understanding necessary for analytical developments [5]. An interesting volume of information about reactor plant events is also available. Models for predicting the unstable behaviour of boiling systems have been set up and qualified by applications ranging from the scale of experimental facilities to the full reactor scale. Both time domain (i.e., transient) and frequency domain (i.e., linearized) models are available to estimate the stability thresholds for even very complex systems.

Natural circulation phenomena have also been thoroughly studied at subcritical pressures, both in single and two phase flow conditions. The relevance of this phenomenon for nuclear reactors is well known, since natural circulation involves several regimes of reactor operation. As previously mentioned, natural circulation and stability phenomena have an extensive intersection in single and two phase flows. A limited list of relevant aspects relating natural circulation at subcritical pressures is reported hereafter:

- Different single and two phase natural circulation modes appear during startup, under normal operating conditions and postulated LWRs’ accidents, as observed in the experiments and code predictions;
- Natural circulation; in particular, passive decay heat removal by natural circulation represents one of the most important features in advanced (GEN III+) and innovative (GEN IV) reactors;
- Different modes of flow instabilities are recognised in natural circulation boiling systems, adding additional phenomena to be addressed to those already observed in boiling channels;
- Vast body of information from experimental and theoretical research on stability and natural circulation performed at subcritical pressures constitutes a useful background also for supercritical water reactor applications, at least in view of the following two main aspects:
  - Knowledge acquired on fundamental phenomena can be helpful in supporting new developments required for supercritical pressure conditions, provided that analogies and differences between the two operating regions (subcritical and supercritical) are properly identified;
  - Part of the operational and perturbed evolution of SCWRs will occur at subcritical pressure, directly involving operating conditions for which previous knowledge is immediately applicable.

2.2.5. Future development

To advance the concepts into the next phase of design optimization, more information specific to their operating conditions are necessary, such as subchannel mixing effect and power profile effect on heat transfer and flow distribution. Heat transfer tests with 3, 4 and 7 rod bundles are completed, and it provided such information for assessing analytical methods like subchannel codes and CFD codes. Full bundle tests will be finally necessary to confirm and demonstrate the designed performance and margin as done for the current WCRs.

2.3. MATERIALS AND CHEMISTRY

In-core and out of core components for SCWRs are required to have sufficient strength and to avoid intolerable degradation or deformation at high temperatures. Major degradation mechanisms of SCWR materials are: general corrosion, environmental assisted cracking such as stress corrosion cracking (SCC) and corrosion fatigue, and radiation embrittlement. Deformation can be induced mainly by void swelling and/or creep. From this point of view, zirconium alloys cannot be used as fuel cladding in SCWRs because these alloys do not have enough strength at high temperatures (i.e., its yield strength rapidly decreases at temperatures above 500 °C) and also because it has very low resistance to general corrosion (i.e., it corrodes easily at temperatures above 400 °C). In order to select SCWR materials, the effects of environment, stress and manufacturing processes need to be assessed.

An extensive experience has already been obtained from R&D on material development and operation of supercritical fossil fuel power plants and nuclear reheated steam cycle systems [30], [31], [32]. The knowledge base can be utilized to develop materials for SCWRs. The selection of cladding materials, which would experience the highest temperature and pressure in the core, has been the focus in material development.

There are basically three directions of materials R&D for SCWRs: one is to select materials among existing ones that are most expected to satisfy requirements of SCWRs (i.e. materials selection), another is to develop or design new materials that are expected to have characteristics better than the existing ones (i.e. materials development) and the third direction between the two is to modify existing materials to improve their weak characteristics (i.e. materials modification) that could be done by coating of the surfaces or shot peening in order to enhance oxidation resistance through cold working of the surfaces. Generally, it takes much longer time (and hence more budget) to develop and qualify a new material than to select or modify the existing materials. Several candidates for cladding material have been identified and assessed against material properties [33]. TABLE 2 lists ranking of these candidates based on material properties, which include corrosion, oxide thickness, stress corrosion cracking (SCC), irradiation assisted stress corrosion cracking (IASCC), creep, void swelling, ductility and strength.

It should be mentioned that the existing R&D methodologies used for current WCRs can be easily applied to the evaluation of SCWR materials. In addition, the experiences of water chemistry strategies in current WCRs and fossil fuel power plants can be used as a starting point for safe and reliable operation of SCWRs.

TABLE 2. RANKING OF CANDIDATE CLADDING MATERIALS FOR SCWRs (reproduced courtesy of Guzonas [33]).

Alloy	Property							
	Corrosion	Oxide Thickness	SCC (un-irradiated)	IASCC	Creep	Void Swelling	Ductility (4% elongation)	Strength
800H	GREEN	YELLOW	YELLOW	YELLOW	GREEN	GREEN	GREY	YELLOW
310S	GREEN	GREEN	YELLOW	GREY	GREEN	YELLOW	GREY	YELLOW
625	GREEN	GREEN	YELLOW	YELLOW	GREEN	GREEN	GREY	GREEN
347	GREEN	YELLOW	YELLOW	YELLOW	GREEN	RED	GREY	GREEN
214	GREEN	GREEN	GREY	GREY	GREY	YELLOW	GREY	RED

- GREEN – Available data suggest that this alloy meets the performance criteria under all conditions expected in the core
- YELLOW – Some (or all) available data suggest that this alloy may not meet the performance criteria under some conditions expected in the core
- RED - Some (or all) available data suggest that this alloy will not meet the performance criteria under some conditions expected in the core
- GREY – There are insufficient data to make even an informed guess as to the behavior in an SCWR core

2.3.1. Major R&D areas

The identification of structural materials for in-core and out of core components and the specification of an appropriate water chemistry control strategy to minimize degradation of the components are two of the major challenges for the development of the SCWR. The main focus is on selection of fuel cladding materials because it is expected that the peak cladding temperatures under normal operation could reach up to 800 °C, which is much higher than those in conventional WCRs.

The need of material related in-pile tests is evident in order to study the effect of irradiation in conjunction with other possible degradation modes affecting materials performance in SCWR service (taking into account e.g. water radiolysis etc.). As an example, higher water flow rates are not achievable in existing water loops connected to the autoclaves in different laboratories worldwide. Development work on in-pile test is ongoing and it is seen as the most important part of international collaboration e.g. through GIF activities.

A test matrix for in-pile tests may be established not only to evaluate the effects of irradiation but also to take into account conclusions and observations from previous laboratory tests such as materials initial condition (effects of surface finish, solution annealed vs cold worked, specimen size etc.). It has been observed that the above mentioned features influence significantly the test results depending on the type of the tests. In addition, reliable data from the material mechanical properties under service conditions (e.g., in order to validate creep models) are needed in order to estimate their long-term operation.

2.3.2. General corrosion and stress corrosion cracking (SCC)

Two different SCWR fuel design concepts have been proposed. The EU [34] and Japan [35] concepts use a freestanding, internally pressurized fuel cladding. New fuel rods would initially be pressurized to about 8 MPa using Helium (He), the internal pressure increasing to about 25 MPa at the end of a cycle due to fission gas release. Internal pressurization imposes requirements on yield strength and creep strength and defines the minimum wall thickness; the maximum allowable corrosion penetration for the HPLWR fuel cladding is 140 µm after 20 000 h (Section 2.1.3.). The Canadian SCWR fuel cladding is designed to collapse onto the fuel pellets, which support the 25 MPa external pressure [36]. The high temperature mechanical strength and creep properties of the alloy become secondary factors in material selection, although fuel swelling due to irradiation will introduce a tensile hoop stress to the cladding. For the Canadian SCWR a maximum allowable corrosion penetration of 200 µm (including oxide penetration along grain boundaries) after 30600 h was specified (~130 µm after 20 000 h, similar to the value specified for the HPLWR), [36].

During reactor operation, oxides will buildup on the cladding surface due to corrosion and deposition of corrosion products originating in the feedtrain (Section 2.1). Small amounts of oxide formation on the cladding surface will increase the roughness and may enhance convective heat transfer from the surface to the coolant. However, significant oxide buildup reduces heat transfer efficiency by conduction through the cladding leading to increases in cladding and fuel temperatures. Therefore a limit on total oxide thickness (oxidation film plus deposited oxide) is required [36].

Data on general corrosion in supercritical water (SCW) are available for more than 90 alloys [2], [37] under a wide range of test conditions and durations. Considerable operating experience exists on the use of stainless steel fuel cladding in LWRs, and the stainless steels Kh18Ni10T and EI–847 were used at the Beloyarsk NPP in superheated steam [2].

Based on currently available data, no candidate material has satisfied properties to resist general corrosion and SCC for fuel claddings of SCWRs. Corrosion behaviours of ferritic/martensitic (F/M) steels, austenitic stainless steels, nickel base alloys, alumina forming steels, and oxide dispersion strengthened (ODS) steels have been studied under SCWR conditions during the past years. Austenitic stainless steels, type 310, 316 and 800H, have been short listed and studied intensively.

In general, the main application for ferritic/martensitic (F/M) steels are the RPV and ex-core components, like piping where temperature is low enough (≤ 400 °C). The austenitic stainless steels are candidates for the internals and fuel cladding. The ODS steels may be an alternative to replace austenitic steels at high operating temperatures but key challenges in manufacturing processes needs to be overcome. Depending on SCWR design, Ni based alloys are problematic to use as core components, since their high Ni content negatively affects core neutronics. Moreover, their creep and oxidation behaviour does not seem to be very good at high temperatures.

Although some of the experimental results based, in particular on SSRT (slow strain rate test) showed that most of the short listed austenitic stainless steels are susceptible to SCC under SCWR operating conditions, the increased susceptibility to SCC is not proven and more sophisticated tests are needed to confirm. Creep is a significant factor that contributes to cracking at temperatures at and above 500 °C for most of austenitic stainless steels and nickel base alloys. Other research points that at even lower temperatures the SCC of Alloy 690 may be expected [38]. In addition, the oxidation rate is too high when considering peak cladding temperatures for alloys with low bulk Cr contents (<18 wt% Cr). It is necessary to understand both the oxidation and cracking mechanisms.

Alumina forming materials, chromium nitride (CrN) and NiCrAlY could be promising candidate materials for surface coating to form a corrosion resistance layer on the fuel cladding. The proposed alumina forming materials have been tested in high temperature water environment at 800 °C. The results showed low corrosion rate and the oxide films are comparatively stable. However, dissolution of alumina under SCW conditions has been observed. CrN and NiCrAlY have showed excellent oxidation resistance at high temperature SCW conditions but stability of the coating layer under cyclic temperature conditions need to be studied further.

Alloy 690 is currently the best candidate material for penetrations on the reactor head. Thermally treated and solution annealed alloy materials were studied under SCWR conditions to understand the effect of oxidation of intergranular carbide on general corrosion and SCC. It shows that under 500 °C, materials with solution anneal condition has lower corrosion rate. Further experiments are ongoing, and more results are expected to clarify the oxidation behavior of intergranular carbides.

2.3.3. Irradiation effects

Compared to other GEN IV concepts, the expected radiation doses for the various proposed SCWR concepts do not differ substantially from those experienced in GEN II and III WCRs, although the range of temperatures at which the materials will be irradiated is much higher. Radiation damaged microstructures lead to significant changes in the material's physical and mechanical properties. The effects such as radiation induced segregation, swelling and creep, H and He generation (from the two step processes: $^{58}Ni(n,\gamma)^{59}Ni(n,\alpha)^{56}Fe$ and $^{58}Ni(n,\gamma)^{59}Ni(n,p)^{59}Co$) have strong temperature dependences; since the cladding temperature varies significantly along the length of the SCWR core, the relative importance of various phenomena may change. For example, material degradation such as grain boundary embrittlement, void swelling, and radiation induced creep and growth may be more of an issue near the core inlet where the cladding temperatures are lower. However, near the core outlet where the cladding temperatures are higher, there will be sufficient thermal activated diffusion of the radiation induced defects to enable recombination of many of the vacancies and self-interstitial atoms leading to annealing (healing) of the damage in this region. The production of He and H can have a pronounced effect on materials performance even at low concentrations; at high temperatures ($> 0.5\ T_m$) He is sufficiently mobile that it can migrate to grain boundaries and cause embrittlement and intergranular fracture with limited ductility, as well as loss of tensile strength. Many of the alloys of interest for SCWR development have been irradiated at temperatures relevant to the SCWR in support of the development of other reactor concepts, e.g., fast reactors.

Irradiation will have effects on general corrosion, environmental assisted cracking (EAC), swelling, mechanical properties. In order to determine the effect of irradiation (in-pile or ion beam, including archived specimens) on material degradation, proton irradiation tests and He effect tests have been done. Some neutron irradiation tests were also done. There are unknown factors such as effects of neutron spectrum and water chemistry. It is known that SCC susceptibility is significantly increased by persistent radiation damage of the metal [39]. To date, irradiation assisted stress corrosion cracking (IASCC) in SCW has not been well investigated.

There has been some progress made on assessment of irradiation effects. However, the data on irradiation effects on material mechanical properties, degradation mechanisms and water chemistry are still limited in the SCWR temperature range. Here remains a 'gap' to be filled.

2.3.4. High temperature mechanical properties

Materials mechanical properties at elevated temperature are extremely sensitive to temperature and represent time dependent properties. Creep is a typical time dependent material property, that is, the special case of inelasticity that relates to the stress induced time dependent deformation under load. The ASME NH subsection pays attention to creep for Class 1 components engineering analysis as well as for an expected loading history. In NH subsection, creep ratcheting phenomena resulted from creep is introduced. At least two mechanisms that high temperature engineering design must take into account are involved in creep ratcheting for materials [40], [41], [42]. First, creep can alter the residual stresses and thus affect the time independent behavior. Secondly, the time dependent deformation can be enhanced because of the nonlinear interaction of primary and secondary stresses.

A significant knowledge gap exists with respect to the influence of the slowly evolving microstructure of stainless steels due to thermal aging. In an SCWR these effects could be significant [32]. Thermal aging is known to create a sensitized microstructure in which chromium carbide ($M_{23}C_6$) precipitates on the grain boundaries, rendering the grain boundaries more susceptible to corrosion. However, the influence of other intermetallic phases has received little attention. Almost all alloys tested to date in support of SCWR development have been tested using as fabricated materials, without much consideration of a slowly evolving microstructure.

Measurements on candidate alloy creep are scarce at SCW conditions. The effect of high temperature oxidation on creep strain rate is also of interest. Clear difference in 316NG and 347H steel behavior between He and SCW environments were noticed [43]. Contrary to this, no environment dependent differences in creep behavior were observed for 1.497 steel or for Alloy 690. Environmental influence on creep rate is thought related to oxidation processes, which probably produce vacancies in the metal/oxide film interface. Repeated tests with longer testing times are required.

The ODS is an effective method in enhancing the strength of base metal while reducing its susceptibility to SCC. Ferritic/martensitic and austenitic stainless steel based ODS steels have been prepared and studied by several institutions, showing promising properties in both mechanical and chemical aspects [44], [45], [46], [47], [48]. However, intensive study is still needed for improving their corrosion performances, and the difficulties in tubing manufacturing and welding are expected to be overcome in the future.

2.3.5. Materials modification and development

The modification of existing materials to improve their oxidation resistance can be done via coating of the surfaces or shot peening in order to enhance oxidation resistance through cold working of the surfaces. Since it takes much longer time to develop a new material than to select or modify existing materials it is well justified to study different types of surface modification techniques.

The surface finish by machining and shot peening has shown beneficial effect on oxidation resistance in past test campaigns. The weight gain of the austenitic stainless steels 316L and 347HFG samples with machined surfaces were smaller than in all other samples with as received conditions. This finding has been confirmed by many corresponding exposure tests at temperatures up to 650 °C showing the beneficial effect of shot peening on oxidation resistance of austenitic alloys [49], [50], [51]. In addition, the effect of surface preparation on the oxidation rate on Ti stabilized austenitic alloy 1.4970 surfaces has been attempted through ion implantation using He^+ and N^+ ions. Further studies are needed in order to confirm the beneficial effect on the oxidation resistance of this material.

An ideal candidate material would be a substrate with good SCC, creep and irradiation resistance, and with a surface layer having high corrosion resistance. To further improve the corrosion resistance without jeopardizing mechanical or irradiation properties corrosion resistant coating may be applied to the surface of the material. Thus coating procedure is seen as another surface modification approach since many mature coating procedures are available for high temperature applications e.g. chemical vapor deposition (CVD), physical vapor deposition (PVD) or thermal spray coatings. The PVD has the advantage over CVD that the deposition temperature can be reduced below 500 °C. In this way, undesired diffusion processes or reactions between substrate and coating are avoided. The PVD process leads to the formation of homogeneous coatings without pores or cracks. The CrN coating could provide an elegant way to protect core internals against corrosion. Preliminary tests have been conducted in SCW conditions at 700 °C indicating very promising results although problems may arise, e.g. with adhesion between the coating and the underlying substrate under cyclic temperatures [52]. However, due to very promising results in terms of oxidation resistance, the effect of irradiation on CrN coatings is underway. Another very interesting coating could be NiCrAlY which has been in extensive use in high temperature steam applications, e.g. in aerospace or industrial gas turbine components [53], [54]. The study on this coating material is under way in many GIF member laboratories and it has showed very promising results up 700 °C in SCW conditions.

Shot peening combined with grain refinement might be the interesting combination due to fact that this has been reported to produce even greater benefit for austenitic alloys (304H and 347HFG) exhibiting oxidation resistance being nearly equivalent to high Cr steels (25% Cr, e.g. HR3C) under test temperature of 650 °C. In general, the results have shown a strong influence of surface finish at the early stages of oxidation. Oxides formed on cold worked or shot peened surfaces were more adherent and much thinner than on a ground surface at temperature between 500−700 °C. The results have indicated that the material surface condition can have a stronger influence than temperature or material composition within the tested ranges. However, in order to confirm this observation much longer exposure times are needed especially for austenitic alloys with lower bulk Cr concentration (< 18 wt%).

2.3.6. Modelling of material degradation and improvement

In order to develop or design new materials, modeling could be beneficial to save cost and time for the experiments. The goal of model development is to use it for extrapolation of materials performance under SCW conditions. This is especially beneficial when taking into account fuel cycles up to 3.5 years. By using modelling tools, the prediction of oxidation resistance of candidate material is possible without extensive long term autoclave tests. Several computer models have been developed to simulate water chemistry and materials degradation. Validation of the models is necessary to apply the models to SCWR conditions.

To achieve lower oxidation rates, structural material performance data have to be obtained and models need to be developed and validated for SCWR conditions. In order to understand the fundamentals of oxidation behavior of materials under SCW conditions, a combination of ex-situ analytical studies (GDOES) of the oxide film forming processes and modelling approaches have been performed during last ten years at VTT in Finland [55], [56]. The overall objective in this modelling effort has been to assess the general corrosion mechanism of candidate materials by using a deterministic model of the oxide layers. As a result of the calculations, kinetics of inner and outer growth and oxide layer restructuring, as well as modifications in the diffusion layer between the bulk substrate and the inner layer have been obtained. Through this course the model has been also verified and validated for several austenitic and ferritic steels, as well as nickel based alloys in SCW conditions.

2.3.7. Radiolysis and water chemistry

In the last ten years the key water chemistry issues for SCWR concepts were identified. Moreover, vast experience from operation of fossil fuel supercritical water plants (FFSCWPs) is available. However, it should be mentioned that water chemistry strategy aims at minimizing corrosion of the feedtrain, rather than the boiler for FFSCWPs. Two key chemistry issues have been highlighted for the SCWR. The first is the transport of corrosion products and impurities such as chloride from the feedtrain to the core. This is still the major pertaining gap in supercritical water technology i.e. the lack of information on the magnitude of the problems of deposition of radioactivity in the external system and of the buildup of internal crud under irradiation. Moreover, it is well known that chloride deposition can lead to failure by SCC. Therefore, the current BWRs operate with feedwater chloride concentrations as low as

0.25 μgkg^{-1}. On the other hand, experience with operation of a pressure tube BWR with nuclear steam reheat channels at Beloyarsk NPP did not show any negative effect with much higher concentration of 25μgkg^{-1} [32]. So far, modeling provides the only means of assessing potential corrosion product deposition under SCWR conditions in the absence of laboratory measurements.

The second issue is the fact that SCWR coolant will be subjected to an intense radiation field as it passes through the reactor core. Water radiolysis reactions resulting from the ionization of water by the passage of gamma and fast neutron radiation (and b-radiolysis close to fuel cladding surfaces and fission fragment radiolysis from tramp uranium) leads to the formation of hydrogen, oxygen and hydrogen peroxide. Although subject to significant uncertainties, some simulations showed that very high concentrations of oxidants are possible in an SCWR core if water radiolysis is not controlled. Of particular concern is the behavior of Cr under oxidizing conditions; the passive films formed on almost all alloys being considered for in-core use in an SCWR are Cr oxides.

2.3.8. Future development

Identification of materials for in core use that are able to withstand the extreme environments, a combination of high temperature, high pressure, irradiation, and rapidly changing chemical properties, has been one of the major hurdles to implementation of this technology. The reawakened interest in the SCWR since the early 1990s, and the international cooperation under the GEN IV International Forum, has led to major advances in SCWR materials and chemistry. A number of viable pressure vessel and pressure tube SCWR concepts have now been proposed [5], and the materials issues have gradually been resolved. While the pressure vessel and pressure tube concepts have unique materials needs (core internals with long in core service lives in the pressure vessel concept; insulated fuel channel in the pressure tube concept), most of the materials issues are generic to both concepts. The selection and qualification of a suitable fuel cladding alloy remains the biggest challenge. While the higher proposed core outlet temperature of the Canadian SCWR concept continues to pose some materials challenges, the adoption of a collapsible fuel cladding for the Canadian concept has significantly reduced the mechanical properties requirements at the peak cladding temperature.

The general corrosion of iron and nickel based alloys under proposed SCWR conditions is now fairly well understood, a reasonable mechanistic framework has been developed, and the remaining gaps in the data required to make reliable extrapolations to the in core residence time of the fuel cladding are being closed.

The SCC behavior of candidate materials is less clear, with sometimes contradictory results being reported by different groups [43], [57], [58], [59], [60]. Additional work is required to understand and reconcile these differences and to define the appropriate testing methodologies. It is clear that adopting the best practices of BWRs and FFSCWPs regarding material selection and feedtrain chemistry control will mitigate in core deposition of impurities such as corrosion products and chloride.

The biggest knowledge gap remains in the behavior of the candidate alloys under irradiation. While the damage experienced by these alloys as quantified by dpa or He production is within the range of that experienced by GEN II and III reactors, the temperature variation across the SCWR core is much larger than found in the current reactor fleet. As a result, the effects of the irradiation on material properties will vary significantly across the core. Little work has been done to understand the coupling of the microstructural changes induced by irradiation with thermal aging effects under SCWR conditions.

2.4. OTHER R&D ACTIVITIES

2.4.1. Safety systems and design evaluations

Safety systems for SCWRs have been investigated specifically per reactor concept. The basic idea of SCWR safety systems is the same as that of conventional WCRs. The emergency core cooling system of SCWRs consists of high pressure and low pressure coolant injection systems and automatic depressurization systems, and some of the concepts adopt or try to adopt passive cooling systems. Most of the SCWR concepts have a pressure suppression type containment vessel like BWRs because SCWRs can have a very small containment vessel due to no steam generators (i.e., direct steam cycle).

In case of loss of coolant accidents, special attention may be paid to two-pass and three-pass cores because it delays refilling and reflooding of the core. Loss of feedwater flow accident also has to be considered as it represents a specific difference compared to conventional WCRs. In a direct cycle SCWR plant, the feedwater works as a reactor coolant. The loss of feedwater flow means that reactor coolant stops to flow.

The effectiveness of safety systems of the SCWR has been designed evaluated with system code analyses as usually is the approach for conventional WCRs, and safety of the concepts has been assessed.

In future, however, the integral tests to validate the system codes are necessary to confirm the effectiveness of safety systems before a prototype/demonstration reactor is designed.

2.4.2. Startup systems and procedures

Starting from cold conditions, the first reactor power will be needed to warm up the steam cycle; as suggested in Ref. [35] either to start with constant supercritical pressure by depressurizing some coolant into a flash tank, or to start with a sliding, subcritical pressure by separating water and steam from the reactor core in external cyclone separators. In either case, the separated liquid is taken to preheat the feed water and the remaining steam is warming up the turbines. As dryout will be unavoidable in the reactor core during subcritical operation, the maximum cladding surface temperature of the fuel rods needs to be checked to avoid damage. In Ref. [2] it is reported about a constant pressure start up and shut down systems for the three-pass core design of the HPLWR, trying to keep the feed water temperature constant to minimize thermal stresses of the reactor pressure vessel. This concept is including also a warm up procedure for the deaerator during start up from cold conditions. A battery of cyclone separators is foreseen outside the containment to produce some steam from depressurized hot coolant of the reactor.

2.4.3. Non-electric applications of SCWRs

The high outlet temperature of SCWR facilitates co-generation of hydrogen using the high temperature steam electrolysis or copper–chlorine cycle. In addition, the high temperature steam is applicable for process heat or oil sand production.

3. SUMMARY OF IAEA ACTIVITIES ON SCWRs

3.1. IAEA PROJECTS AND ACTIVITIES ON SCWRs

The IAEA started its activities on SCWRs in 2006 and launched the first project in 2008. Since then, the IAEA has implemented several projects to foster collaboration in R&D and information exchange among interested institutions in Member States.

The goal of the IAEA projects on SCWRs is to facilitate technology development in Member States mainly by:

— Providing forums for researchers and engineers to collaborate in R&D related activities;
— Promoting information exchange of advanced technology for SCWR development;
— Educating newcomers who are interested in science and technology of SCWRs.

3.1.1. Coordinated research projects

In support of Member States' efforts, the IAEA started, in 2008, a Coordinated Research Project (CRP) on *Heat Transfer Behaviour and Thermo-hydraulics Code Testing for Supercritical Water Cooled Reactors (SCWRs)*. The two key objectives of this CRP were to establish a base for accurate data related to heat transfer, pressure drop, critical flow, natural circulation and flow stability of supercritical fluids and to test analytical methods for SCWR thermal hydraulics behaviour and identify code development needs. In total, 16 institutes from 10 Member States and two international organizations were involved in this CRP, and close collaboration in several tasks resulted in technology advancement. The CRP was completed in September 2012. Information generated from this CRP was documented in an IAEA Technical Document [5] and numerous publications and reports. By the end of 2016, a database of thermal hydraulics parameters of interest to SCWR development was compiled and housed in the central server of the OECD Nuclear Energy Agency (OECD/NEA), based on the special agreement with the IAEA. It is accessible to the CRP participants only for future collaboration.

Despite the completion of the CRP, several collaborative initiatives continued among institutions that participated in the CRP. Most of these institutions expressed their strong interest in, and support for, initiating a new CRP on thermal hydraulics of SCWRs to continue the momentum of international collaboration. Accordingly, the IAEA launched, in October 2014, a new CRP entitled *Understanding and Prediction of Thermal Hydraulics Phenomena Relevant to Supercritical Water Cooled Reactors*. The objective of this new CRP was to improve the understanding and prediction accuracy of thermal hydraulics phenomena relevant to SCWRs and to benchmark numerical toolsets for their analyses. The identified scope of collaboration was considered to cover applied research and development (R&D), as compared to the basic R&D carried out under earlier CRP. Twelve (12) institutions participated in this CRP from 10 IAEA Member States, and the OECD/NEA was in cooperation to host a database of experimental and analytical results contributed from the CRP participants. The expected outcomes from this CRP included (i) enhancement in understanding thermal hydraulics phenomena, (ii) sharing experimental and analytical results, and prediction methods for key thermal hydraulics parameters, and (iii) cross training of personnel between participating institutions through their close interactions and collaborations.

3.1.2. Technical meetings

As a complement to the CRPs in the area of thermal hydraulics for SCWRs, Member States expressed an interest in holding a technical meeting at which specialists in the areas of SCWR system design and thermal hydraulics would have the opportunity of participating in extended technical discussions on science, technology and engineering of SCWRs. The purpose of these technical meetings was to provide a platform for detailed presentations and technical discussions on recent progress in R&D activities related to SCWRs, leading to the exchange of results and fostering worldwide collaboration in further R&D activities. A further goal of the meetings was to review and update the status of scientific and engineering related knowledge underlying the SCWR concepts. The programmes included summary sessions to enable participants to contribute to the summary and highlights of the meeting and to make recommendations to the IAEA on future activities in this field.

The IAEA have organized two technical meetings on Heat Transfer, Thermal Hydraulics and System Design for SCWRs in collaboration with the CRP participants and other experts from Member States.

The first technical meeting on thermal hydraulics for SCWRs was hosted by the Government of Italy through the University of Pisa, and was held in Pisa, Italy, from 5–8 July 2010. The meeting was attended by 54 participants from 17 Member States. The second technical meeting was hosted by the Government of the United Kingdom through the University of Sheffield, and took place in Sheffield, UK, from 22–24 August 2016. In total, 33 participants from 9 Member States and the host organization participated in the meeting.

In parallel to the Technical Meetings on thermal hydraulics for SCWRs, three technical meetings on *Materials and Chemistry for SCWRs* have been organized. The first technical meeting was hosted by the Joint Research Centre of the European Commission, and was held in Petten, the Netherlands, from 18–22 July 2011. It was attended by 24 experts from 13 Member States and the host organization. The second one on the same topic was hosted by the Government of China through the Nuclear Power Institute of China, and was held in Chengdu, China, from 22–26 July 2013. It was attended by 20 participants from 6 Member States and the host organization. The third technical meeting was hosted by the Government of Czech Republic through the Research Centre Řež, and was held in Řež, Czech Republic, from 10–14 October 2016. In total, 19 participants from 9 Member States and two international organizations participated in the meeting.

3.1.3. Training courses

As a mean of disseminating knowledge generated and collected through the work performed under the CRPs and through the technical meetings, the IAEA developed content to be used for courses on Science and Technology of SCWRs. These courses provided a comprehensive and up to date review of the science and engineering of SCWR concepts, including thermodynamics, thermal hydraulics and heat transfer, neutronics and core designs, materials and chemistry, system design and safety, and a description of various SCWR concepts currently under development in the world.

The first course was organized by the IAEA together with the International Centre for Theoretical Physics (ICTP), and it was held at ICTP, Trieste, Italy, June 27–July 1, 2011. A total of 22 international participants from 18 Member States, ranging from graduate students to university professors and from engineers to regulators, attended the course. The additional three courses were organized by universities in cooperation with the IAEA, and they were held at the McMaster University in Hamilton, Canada, 16–21 July 2012, at the Shanghai Jiao Tong University in Shanghai, China, 26–30 August 2013, and at the University of Ontario, Institute of Technology (UOIT), Oshawa, Canada, 4–8 July 2016.

3.1.4. Cooperation in international meetings

The IAEA has cooperated in several International Symposia and Meetings on SCWRs, including the International Symposium on SCWRs (ISSCWR) since 2011 and GIF SCWR Information Exchange Meetings since 2012.

3.2. HEAT TRANSFER, THERMAL HYDRAULICS AND SYSTEM DESIGN FOR SUPERCRITICAL WATER COOLED REACTORS

The Technical Meeting on Heat Transfer, Thermal Hydraulics and System Design for SCWRs was held in Sheffield, UK, 22−24 August 2016. It was the second meeting focusing on the following main objectives, with an emphasis on application and design issues:

- Review and discuss progress in the development of experimental correlations, theoretical equations and numerical methods to describe the thermal hydraulics behaviour of fluids under supercritical pressure conditions;
- Evaluate comparisons of analyses and numerical predictions of thermal hydraulics codes against theoretical estimates and experimental data for supercritical pressure fluids;
- Review the current status of SCWR concepts and their core design, system design and approach to safety.

In total, 24 nominated participants from 9 Member States and 9 observers from the host organization participated in the meeting.

The technical meeting consisted of the Opening Session, 3 Technical Sessions, 2 Summary Sessions and the Closing Session. Thirteen presentations were provided in the Technical Sessions, followed by very active technical discussions, and several general issues related to the session topic were discussed at the end of each Technical Session. Based on the presentations and discussions at the Technical Sessions, the current status of R&D and remaining challenges were discussed in the area of thermal hydraulics for SCWRs at one of the Summary Sessions, and future collaborative activities were proposed and discussed.

3.2.1. SCWR conceptual design and thermal hydraulics

The first Technical Session was focused at the concept development of SCWRs and thermal hydraulics issues related directly to the designs development. Three presentations focused on the development status of SCWR concepts in Canada, Euratom and Russian Federation, and relevant thermal hydraulics issues:

(a) Thermal-Hydraulics Studies of the Canadian SCWR Programme
Laurence K.H. Leung, Canadian Nuclear Laboratories (CNL), Canada

Canada is developing a pressure tube type SCWR concept with enhanced safety, proliferation resistance and sustainability characteristics at compatible costs to current fleet of nuclear reactors. It joined the GEN IV International Forum for cooperative R&D with the international community and is currently collaborating with researchers in China, European Union, Japan and the Russian Federation in developing SCWR concepts. Despite of differences in the core configuration of SCWR concepts being pursued in various countries, technical issues encountered in several technology areas are common and can be addressed collectively.

Thermal hydraulics at supercritical pressures has been identified as one of the critical technology areas in the development of the SCWR concepts. It affects the operating power and the safety margin of the SCWR concept, and also has an impact on the selection of cladding material and neutronic design. Canada has initiated a national programme in support of R&D for the SCWR. Thermal hydraulics studies have been performed at Canadian Nuclear Laboratories and Canadian universities. A number of joint projects was established to extend the thermal hydraulics R&D effort through bilateral collaborations between Canadian Nuclear Laboratories and researchers in China, the Russian Federation and the United Kingdom. In addition, Canada's participation in the Cooperative Research Project hosted by the IAEA has facilitated further enhancement of the experimental database and the analytical tools.

A summary of the thermal hydraulics studies in the Canadian SCWR programme was presented. Heat transfer experiments performed with tubes, annuli and bundles in water, carbon dioxide or refrigerant flows are described. Experimental data obtained from these experiments have been examined for investigating separate effects on heat transfer (such as diameters, spacing devices, etc.). Analytical tools have been assessed against the latest experimental data for quantifying the prediction capability and applicability. These assessment results were presented.

(b) The Mutual Influence of Thermal-hydraulics and Materials on the Design of SCWR−Review of the Results of the Project "HPLWR Phase 2"
Joerg Starflinger, University of Stuttgart, Germany

Core design of SCWR is strongly influenced by the variation of thermodynamic properties of supercritical water near the critical point. Nonlinearity of the property changes with temperature provides a special challenge in a hot channel analysis.

Applying the hot channel factor analysis to the expected heat up of a SCWR core resulted in the three-pass core design as analyzed in the HPLWR Phase 2 project. The three passes resulted from necessity to mix the supercritical fluid during the heat up inside the core to avoid hot streaks, which could challenge the cladding material. For the HPLWR project, a maximum cladding surface temperature of 630 °C was selected. Three-pass simply means that the flow passes the reactor core three times: upward flow in the core mixing in an upper plenum downward flow in a different section of the core mixing in a lower plenum upward flow in a different section and delivery of supercritical water of 500 °C to the turbine.

In the HPLWR Phase 2 project, wire wrap spacers were foreseen instead of grid spacers that are used in BWR and PWR. Such wire wrap spacers mix quite well, but could possibly lead to hot spots on the cladding surface, which must be avoided though a suitable design.

Although the flow is well mixed within an assembly, the non-uniform neutron flux (and power) profile across an assembly still results in a non-uniform heat up of the supercritical fluid. In the HPLWR design, nine smaller assemblies were grouped into one assembly cluster with common head and foot piece. The grouping was selected because control rod devices from existing PWRs (including control rod drives) may be used and the allowed number of flanges in the closure head of the reactor was limited at the increased pressure. However, for flattening of the neutron flux and power profile, smaller fuel assemblies (like in BWRs) would be more appropriate, which are also more flexible for shuffling.

(c) Heat Transfer Problems in Fuel Assembly of VVER-SCP
A.Y. Kuzmichev, OKB "GIDROPRESS", Russian Federation

Three concepts of reactor plant (RP) with supercritical water coolant VVER-SCP (SCWR), which refer to the GEN IV systems, were developed in Russia:

- VVER–SCP–1700: one circuit high power RP, with fast resonant neutron spectrum in the core;
- SPCS–600: two circuit RP, with pseudo vapor supercritical primary coolant and fast neutron spectrum in the core;
- VVER–SCP–I: two circuit RP, with supercritical water coolant in the primary circuit and regulated neutron spectrum.

Up to the moment, OKB "GIDROPRESS" implements engineering development of VVER-SCP-1700 and performs thermal hydraulic analyses.

Several problems, which were discovered during the subchannel calculations of the coolant flow in fuel assembly of one-circuit VVER-SCP-1700 must be studied with the first priority.

The report emphasized the problem of the displacement of coolant from the more heat-stressed part of the bundle to the less heat stressed. The results of the calculations concerning this problem were presented. New specification of the future thermal hydraulic subchannel calculations in the framework of development and verification of TEMPA-SC code was presented.

3.2.2. Heat transfer and thermal hydraulics for SCWRs

In this session, heat transfer and other thermal hydraulics phenomena of supercritical fluids were discussed in detail. Five presentations were provided focusing at heat transfer to supercritical pressure fluids, the onset of heat transfer deterioration, deteriorated heat transfer, fluid to fluid scaling of heat transfer, near wall turbulent structure, and thermal acceleration effects.

(a) Effect of Tube Diameter on Supercritical Heat Transfer to CO_2
Nathan Kline, University of Ottawa, Canada

Extensive convective heat transfer measurements have been collected in the Supercritical University of Ottawa Loop (SCUOL) with carbon dioxide at supercritical pressures flowing vertically upwards in tubular test sections having inner diameters equal to 22.0, 8.0, and 4.6 mm. Outer wall temperature was measured by a large number of thermocouples, from which the inner wall temperature and the local heat transfer coefficient were estimated. The measurements extend over wide ranges of conditions, which cover both the normal and deteriorated heat transfer modes.

Experiments were conducted using all three test sections with the purpose of determining the tube diameter effects on the local heat transfer coefficient for normal heat transfer. Measurements were collected for 15 different flow and heating conditions, all at $P/P_{critical} \approx 1.13$. The results showed that, in most cases, the corresponding local heat transfer coefficient was higher for smaller diameter tubes. Nevertheless, for flows at low mass fluxes ($G \leq 400$ kg/m^{2}s), the local heat transfer coefficient values from the 8 and 22 mm test sections were comparable, especially at high specific bulk enthalpies ($H_b >$ 300 kJ/kg); for $G = 300$ kg/m^{2}s, the local heat transfer coefficient values from all three test sections in the high enthalpy range actually nearly coincided. At higher mass fluxes, however, the local heat transfer coefficient values remained distinct for each test section over the full range of H_b values examined, with higher local heat transfer coefficient values for smaller test section diameters. To determine the conditions at the onset of heat transfer deterioration (HTD), experiments were also performed by gradually increasing the wall heat flux q, while keeping the pressure constant at $P \approx 1.13P_{critical}$, the mass flux constant in the range 200 kg/m^{2}s $\leq G \leq$ 1000 kg/m^{2}s and the inlet temperature T_{in} constant in the range 0 °C $\leq T_{in} \leq$ 35 °C, until a temperature spike was observed in the wall temperature profile. It was found that, at the onset of heat transfer deterioration (HTD), the heat flux limit q^* could be fitted by a power law of G with the same exponent for all three test sections. The heat flux limit for the 8 mm test section, used as a reference, could be represented as $q_{ref}^* = 6.2 \cdot 10^{-5} G^{2.2}$. The proportionality constant for the 4 mm tube was essentially the same, whereas for the 22 mm tube it was significantly lower.

(b) Development of Correlation for Heat Transfer Enhancement and Deterioration for Supercritical Fluid Using Freon 22 Experimental Results
S.K. Dubey, Atomic Energy Regulatory Board, India

A Supercritical Freon Test Facility was designed with R22 as working fluid and built with two vertical tubular test sections of the inner diameters equal to 6 mm and 13.5 mm, respectively. Experiments with vertically upward flow at 55 bar system pressure were carried out and a thermal camera was used to obtain wall temperatures every 1−1.5 mm along the axial direction.

Initial experiments were conducted to demonstrate the reduction in peak HTE with increase in heat flux. As low heat flux is needed to carry out experiments for HTE, the rise in bulk fluid temperature across the test section will be very low especially near the pseudocritical temperature and therefore, experiments were carried out in a piece wise fashion. Experiments were performed by changing the inlet temperature with the help of a preheater and using a sufficient overlap between experiments, the results were joined together. The results showed that the HTE is independent of inlet temperature and the data could be properly integrated together to obtain the behaviour over the required length.

Experiments were then conducted for HTD at several heat and mass flux values and inlet temperatures. It was observed from experimental results that the onset of HTD occurs when q/G is more than 0.056 to 0.072 kJ/kg and when the inlet temperature was lowered; the onset of HTD appeared at a relatively higher *q/G*. The bulk fluid enthalpy and temperature at which onset of HTD appeared also reduced when the inlet temperature was decreased. A sharp rise in the wall temperature was initiated when the wall temperature was just above the pseudocritical temperature at relatively high heat flux values. Two peaks in wall temperature were observed in the results at the lower inlet temperature. It was observed that that bulk fluid enthalpy at which peak HTD occurred decreases with increase in heat flux. The heat transfer coefficient values in the region of the test section before the start of HTD were in good agreement with the predictions using the Dittus–Boelter correlation and an appreciable recovery of heat transfer coefficient after the deterioration was observed at relatively low heat fluxes in the HTD zone.

A new non-dimensional number has been identified and the criteria for onset of heat transfer deterioration is proposed, and separate correlations to predict heat transfer coefficient for HTE and HTD have been developed based on the experimental results, for example, HTC correlation is developed for reduction in heat transfer enhancement. The correlations are based on non-dimensional numbers and therefore can be applied to any supercritical fluid. These correlations were used for reported data employing water and CO_2, and the comparisons were noticed to be satisfactory.

(c) Fluid to fluid scaling of heat transfer with supercritical pressure fluids: recent considerations and future perspectives
Andrea Pucciarelli, University of Pisa, Italy

The contribution reported on the ongoing work at the University of Pisa about fluid to fluid scaling with supercritical pressure fluids. The first studies in this regard started in 2006 when relevant non dimensional quantities were introduced for analysing flow instabilities, supposing the existence of similarities between the exceeding of the pseudo critical temperature in supercritical pressure fluids and the boiling threshold in subcritical pressure ones. Since the beginning interesting features where noticed, e.g. the fact that the non-dimensional density trend (ρ^*) seems to be a function of the dimensionless enthalpy (h^*) only, regardless of the supercritical pressure and, in general, of the fluid. This feature allowed the research team in Pisa to successfully find the similarities between dynamic behaviour of different supercritical fluids in forced and natural circulation. In fact, as the $\rho^* - h^*$ trend is similar for different fluids, once the same h^*_{in} and the same Δh^* along the same heated length (called N_{TPC}) are imposed, the fluids may undergo similar buoyancy effects. In particular, the same h^*_{bulk} and $\rho^*_{bulk}$are expected in correspondence of the same axial position. While obtaining success for dynamic behaviour, mainly driven by bulk fluid effects, difficulties were found when facing heat transfer phenomena. In fact, incompletely coherent behaviours were noticed when comparing different fluids adopting the recipe that proved to be suitable for studying instabilities. In particular, carbon dioxide tended to return stronger heat transfer deterioration phenomena than the ones reported by water once similar boundary conditions were set.

The main reason of this discrepancy was due to the fact that, though for global flow dynamics by imposing Δh^* is sufficient for defining the density differences which generate the leading forces of the phenomenon, in heat transfer phenomena the local value of heat flux becomes very important, since different fluids show different heat transfer capabilities. In particular, this happens because, unlike density, there is no close similarity between the dimensionless trends of specific heat, thermal conductivity and dynamic viscosity when considering different fluids.

In the frame of the latest works the problem was solved, at least basing on CFD data, with the help of RANS calculations. Instead of imposing the same Δh^* along the same heated length, a different length is defined scaled through the values of the Stanton number. Though the same N_{TPC} is maintained, no geometrical similarity in the *L/D* ratio exists anymore and now the same h^*_{wall} value is expected when the same h^*_{bulk} conditions are obtained. This technique was tested over a small number of selected cases adopting four different supercritical fluids. The analyses returned promising results; very similar h^*_{wall} trends were obtained and similarities were also noticed for the dimensionless velocity, enthalpy and turbulent kinetic energy radial trends.

Unfortunately, no experimental evidence exists at the moment for the proposed scaling method. This is mainly due to the fact that experiments adopting carbon dioxide as working fluid often consider very near critical conditions, corresponding to bulk temperature values up to 350 °C for water, which certainly deserve particular and expensive facilities for being carried on. Consequently, new ad hoc experimental campaigns may be planned for proving whether or not the proposed technique can be relied upon.

With the aim of obtaining more realistic results, analyses are currently being performed at the University of Pisa by adopting LES techniques and are summarised in the present paper. Carbon Dioxide is considered for the reference case and scaling attempts adopting water are currently running. The proposed results are not definitive yet; however, some interesting features can be observed: the calculations return quite similar trends though, due to the low Reynolds number of the selected case, fully developed turbulent flow is not assured.

A parallel work, carried on in cooperation with the University of Sheffield, takes into account DNS techniques for performing the scaling attempts. RANS calculations were performed in order to plan the DNS campaign by selecting suitable boundary conditions. Three scaling cases were initially planned, nevertheless, at the moment, only one seems having reached sufficiently stationary conditions. The results obtained in the mentioned case are here presented and commented.

Many analyses are then still needed, nevertheless the obtained results seem to be promising. The main goal of the present contribution is then trying and show the effectiveness of the proposed scaling technique and, moreover, to persuade researchers to try and consider the proposed method as a cue for performing experimental campaigns.

(d) Role of Near–Wall Turbulent Structures in the Heat Transfer of Transcritical Channel Flow
Kukjin Kim, Purdue University, USA

Paper summarized a series of direct numerical simulations of a canonical channel flow setup to characterize the turbulent heat transfer effects in a transcritical thermodynamic regime for tetrafluoroethane (R−134a). To unambiguously quantify the turbulent heat transfer, the top and bottom channel walls are maintained at different isothermal temperatures based on a selected temperature difference above (top wall) and below (bottom wall) the pseudo boiling (PB) of the fluid (determined based on the bulk pressure of the channel). This setup allowed to study the thermodynamic region in which the fluid undergoes the greatest thermophysical variation in a statistically steady state. The channel flow was simulated using a high order compact finite difference scheme (6^{th} order accuracy) for the solution of the conservative Navier–Stokes equation. The equations are closed with a cubic Peng–Robinson equation of state and Chung's model was used to account for the thermophysical properties of the fluid. A rigorous grid convergence study is undertaken to show that characteristic point to point oscillations emerging from the flux calculations (for conservative schemes) can be bounded with sufficient spatial resolution. Therefore, a high order simulation, which fully resolved all scales of turbulent motion, could be attained without additional artificial dissipation, limiters or non-conservative formulations.

Turbulence was the primary contributor to the near wall heat transfer and its effect is modulated by the underlying nonlinear thermodynamics. Time averaged density and temperature profiles showed the highest thermodynamic gradients are located near the walls with a qualitatively symmetric distribution about the centerplane. The root mean square of the density has typical peaks in the near wall region. As the temperature difference between the top and bottom wall increases, so does the asymmetry of the density fluctuations. Enhanced fluctuations are observed near the cold wall with increasing temperature difference, but more importantly, it should be noted note the emergence of a distinct centerplane peak. A similar anomalous behaviour is observed in the total enthalpy flux in the streamwise direction. Here it should be noted a discontinuous profile of the enthalpy flux near the centerline when ΔT increases. Near wall turbulent structure visualizations reveal that the near wall heat and momentum transfer is primarily governed by the ejection of dense fluid lumps away from the cold wall (bottom) towards the center of the channel. This ejection process occurs as streamwise aligned vortical structures lift up the high density fluid (below the PB point), and given its increased momentum, these lumps migrate to the center of the domain where the elevated temperature results in a pseudo-transition away from the walls. This mechanism leaves its mark on the instantaneous temperature gradients at the wall. This structural understanding in transcritical flow provides new insight into the mechanisms of heat transfer in these complex thermodynamic states. This work will be continued with the objective of developing relevant wall models for the accurate modelling of turbulent heat transfer in transcritical flows.

(e) Thermal Acceleration of SCW Flow in Heat-Generating Channels as a Factor of Heat Transfer Deterioration
V. Razumovskiy, National Technical University of Ukraine (NTUU), Ukraine

Numerous experimental studies of heat transfer and pressure drop in smooth tubes cooled by SCW revealed that at certain high heat flux to mass flux ratios (q/G) and coolant temperature ranges from about 340 to 400 °C, i.e., in the region of transition from liquid to gaseous state, hydraulic resistance to thermal acceleration of the flow, Δp_{ac}, could exceed friction resistance, Δp_{fr} (case of high acceleration condition). At temperatures below and above this range, the total pressure drop Δp is a function either of a very strong dependence of coolant viscosity upon temperature (viscous flow, if $t_f < 150$ °C) or of viscous and inertial forces, if 150 °C $\leq t_f \leq$ 340 °C or $t_f \geq$ 400 °C.

To get correct values of acceleration resistance factor ξ_{ac} the IVTAN method of two pressure drops tested at supercritical carbon dioxide was applied for SCW. These experiments were conducted at flow enthalpies within the range from 1370 to 2180 kJ/kg covering transition from liquid to gaseous state.

The analysis of experimental data revealed that heat transfer deterioration occurs when $\xi_{ac}/\xi_{fr} > 1$ under negligible effect of buoyancy. This is attributed to the laminarization of boundary layer and resulted in suppression of turbulence. Thus, these data could serve as an additional evidence of the crucial effect of thermal acceleration on heat transfer and its deterioration under the conditions of the developed turbulent flow even without impact of free convection.

A correlation was proposed for the boundary heat flux to mass flux ratio $(q/G)_{\mathrm{b}}$ for the transition from normal to deteriorated heat transfer (DHT). It was based on 120 experimental tests that covered a range of q/G, from 0.3 kJ/kg to 1.5 kJ/kg. The correlation was in a good agreement (within ±25%) with the results obtained for one 1 rod (annular channel), 3 rod and (tests are still in progress), and 7 rod bundle simulators (about 200 experimental values of $(q/G)_{\mathrm{b}}$). Unlike other correlations, it took into account: (a) an initial thermal state of coolant in relation to it at the point of maximal isobaric specific heat capacity; and (b) the length of heated section from inlet to the initiation point of DHT. In many cases, the DHT boundary are almost twice the predictions using current available correlations. In addition, it has been observed, when $(q/G) > (q/G)_{\mathrm{b}}$, a slight change in operating conditions could lead to a sharp variation in wall temperature (exceeding 50 °C–100 °C).

3.2.3. Numerical simulation and code development for SCWRs

This session discussed numerical analysis and development of numerical simulation codes for SCWRs. Five presentations were given on simulation of transient heat transfer, CFD benchmark analysis, DNS, and code developments for design and analysis.

(a) Simulation of Transient Heat Transfer in an SCWR Fuel Assembly Test at Near Critical Pressure
Thomas Schulenberg, Karlsruhe Institute of Technology (KIT), Germany

While supercritical water is a perfect coolant with excellent heat transfer, a temporary decrease of the system pressure to subcritical conditions, either during intended transients or by accident, can easily cause a boiling crisis with significantly higher cladding temperatures of the fuel assemblies. Such situation is planned to be tested with a small fuel assembly of four rods, to be operated in a critical arrangement with supercritical water inside a research reactor in the Czech Republic. First out-of-pile tests of this experiment have recently been performed in the SWAMUP facility at SJTU in China.

Some of the transient tests have been simulated as shown in Ref. [61] with one dimensional MATLAB code, assuming quasi steady state flow conditions, but time dependent temperatures in the fuel rods. Heat transfer at supercritical and at near critical conditions was modelled with a recent look up table [62] and subcritical film boiling was modelled with the look up table as described in Ref. [63] giving the best accuracy at minimum run time of the code. Moreover, a conduction controlled rewetting process was included in the analyses, which is based on an analytical solution provided in Ref. [64]. This fast method may be applicable later to any system code for nuclear application.

The new method could well reproduce the boiling crisis during depressurization from supercritical to subcritical pressure including rewetting of the hot zone within some minutes, but the peak temperature was somewhat under predicted. Tests with a lower heat flux, which did not cause such phenomena, could be predicted as well. In another test with increasing pressure, however, a boiling crisis was also observed at a heat flux, which was significantly lower than the critical heat flux predicted by the CHF look up table provided in Ref. [65].

(b) CFD Benchmark Analysis for 2×2 Fuel Rod Bundle
Kate Lyons, University of Wisconsin, Madison, USA

In order to better understand and adequately predict heat transfer in supercritical fluids, it is important to complete simulation benchmarks to compare the model results with experimental data. This benchmark process allows for the simulation model to be tuned to the fluid and geometry application to enable better future prediction capability of the model. As part of the IAEA CRP on SCWR thermal hydraulics, CRP participants validate their CFD or subchannel code with four rod bundle heat transfer experimental data. Data is being provided by both Shanghai Jiao Tong University (SJTU) and the University of Wisconsin, Madison (UW Madison). In this study, progress towards CFD model validation against SJTU experimental data is presented.

In the test section of the UW Madison facility flow loop, water flows upwards over four heated rods which simulate a two by two fuel rod bundle with a pitch to diameter ratio of 1.33. The rods are manufactured to create a cosine axial heat flux profile to better simulate reactor like conditions. Thermocouples were swaged under the cladding of the heater rods at various axial and radial positions to provide temperature data for the system. The UW Madison test section was simulated using the CFD program Fluent under subcooled water conditions to verify geometry and to establish an initial working model. These initial simulations were completed before extending the simulation domain to supercritical water and modelling the SJTU test section.

Many experiments have been completed using the SWAMUP test facility flow loop at SJTU to study heat transfer characteristics in supercritical and subcooled water. In the SJTU test section, four tubes are uniformly heated within the square flow channel with a pitch to diameter ratio of 1.18. There are six equally spaced grid spacers that hold these tubes in place. Each heated tube contains a sliding thermocouple that allows for temperature measurement along the axial length and at defined radial positions.

Two supercritical water experimental cases from the SJTU facility were chosen to model using CFD and the results were compared. Case A had an inlet temperature of 346.4 °C and an inlet pressure of 25.09 MPa with a mass flux of 844.52 kg/m^2/s and a surface heat flux of 800.2 kW/m^2 per heater rod. Meanwhile, Case B had similar inlet temperature and pressure conditions of 340.1°C and 25 MPa respectively with considerably lower mass flux and heat flux conditions of 451.2 kg/m^2/s and 551.6 kW/m^2 per rod respectively. Different simulation models were used for both cases and their results were compared to the experimental data. This comparison led to a better understanding of the requirements of a simulation model for use in supercritical fluids. Furthermore, simulation of the two experimental cases allowed for comparison of differing heat and mass flux on the modelling technique.

The simulation techniques developed during this part of the benchmark will be used to predict the heat transfer characteristics of the UW Madison test section at supercritical conditions. These models will need to be able to provide predictions regardless of heat flux profile and pitch to diameter ratios. By performing these benchmarks between facilities, the resulting simulation techniques will provide much useful information to further the understanding of supercritical heat transfer phenomena.

(c) Direct numerical simulation of fluid flow at supercritical pressure in a vertical channel
Wei Wang, The University of Sheffield, Sheffield, UK

Fluids at a supercritical pressure are accompanied by strong variations of thermodynamic and thermophysical properties, buoyancy influences and abnormal thermal developments, which have complex effects on turbulence and heat transfer, making predictions and modeling of such flows a difficult task. A direct numerical simulation of water at supercritical pressure in a vertical channel is carried out to study effect of variation of thermo-properties and buoyancy on turbulence and heat transfer. The two walls of a vertical channel flow are set to be at constant but different temperatures to isolate effects of variable properties and buoyancy from intricate thermal and flow developments. Under this condition, heat input from the heating wall and heat removal from the cooling wall are balanced to finally achieve a fully developed state with no heat advection but only thermal diffusion statistically. A new perspective on variable property and buoyancy effects is proposed, in which the phenomena are decomposed into bulk effect of variable property, density magnitude effect, density gradient effect, buoyancy effect interacted with density magnitude and buoyancy effect interacted with density gradient. The importance and impact mechanism of each element are compared and discussed. Turbulent and thermal characteristics near the cold and hot walls are discussed with regard to velocity, temperature, turbulence heat flux, turbulence statistics and turbulence productions. Turbulent flow structures are also investigated via the Lumley triangle analysis and isosurface visualization of instantaneous flow structures.

(d) Development of Three Dimensional Code Systems for SCWR Core
Lianjie Wang, Nuclear Power Institute of China (NPIC), China

In supercritical water cooled reactor (SCWR), huge change of the coolant density changes the core slowing down cross section field and has an important influence on the power distribution, and forms the complicated coupling characteristic between neutronics and thermal hydraulics. The coupling between neutronics and thermal hydraulics must be considered in core steady design, and also the spatial power distribution and its change must be considered for core transient analysis and safety evaluation.

The paper mainly focuses on the study of coupling three dimensional neutronics and thermal hydraulics simulation for SCWR core steady state analysis and transient analysis. Coupled three dimensional core steady state analysis code and transient analysis code are developed and preliminarily validated.

The coupled three dimensional neutronics/thermal hydraulics code SNTA (SCWR coupled neutronics/thermal hydraulics Analysis code) is developed for SCWR core steady state analysis. The cross section fitting module is improved for SCWR, and the in-core fuel management code NGFMN_S is developed based on three dimensional Nodal Green's Function Method. NGFMN_S and subchannel code ATHAS is then modularly coupled, and the appropriate outer iteration coupling method and self adaptive relaxation factor are proposed for enhancing convergence, stability and efficiency of coupled N/TH calculation. By comparing the existing steady state code for SNTA verification, the numeric results show that SNTA is accurate and efficient for SCWR core steady state analysis.

Then the coupled three dimensional neutronics/thermal hydraulics code STTA (SCWR Three dimensional Transient Analysis code) is developed for SCWR core transient analysis. Nodal Green's Function Method NGFMN_K is used for solving transient neutron diffusion equation. The SCWR subchannel code ATHAS is integrated into NGFMN_K through the serial integration coupling approach. The dynamic link libraries method is proposed for coupling computation for SCWR multi flow core transient analysis. The reliability and applicability of STTA are well proved by the PWR benchmark problem and SCWR rod ejection problems.

(e) Thermal hydraulics code development in SJTU for SCWR
Xiaojing Liu, Shanghai Jiao Tong University (SJTU), China

Due to its potential for high thermal efficiencies and considerable plant simplifications, SCWR have attracted strong interests of the international nuclear community and is recommended as the only WCR among the six GEN IV concepts. In the recent years, the application of supercritical fluid arouses lots of attention in the nuclear R&D field due to its application to the SCWR. As one of the most important research institutes studying SCWR, Shanghai Jiao Tong University (SJTU) plays an important role in the supercritical fluid field. In the present paper, fundamental studies, both experimental and numerical, on heat transfer are presented, to provide basic information for understanding the heat transfer mechanisms and to develop methods for code development for safety analysis of SCWR. The main achievements can be summarized as follows:

- Perform CFD analysis for the supercritical fluid in subchannels, and develop new turbulence model to improve the predictive capability for heat transfer and mixing behavior;
- Improve some key models in subchannel code for supercritical fluid, e.g. mixing, heat transfer, pressure drop, and carry out the validation work by experimental data;
- Improve the current codes and perform the safety analysis for the current SCWR design by using the system code and subchannel code, to demonstrate the feasibility of the passive design system in SCWR;
- Couple the system code and subchannel code and perform the safety analysis for the SCWR fuel qualification test (FQT) facility.

The results achieved in this project can contribute to understanding the basic thermal hydraulic phenomena and to improve the accuracy of the current prediction method and SCWR design.

3.2.4. General discussion on thermal hydraulics for SCWRs

Recent studies continuously illustrated the complexity of the thermal hydraulics for supercritical fluids, especially heat transfer at the deteriorated region. Experimental data obtained in tubes and bundles provided additional insight to the phenomena and facilitated improved understanding. Correlations have been derived to improve the prediction accuracy of the thermal hydraulics parameters.

The assessments of CFD tools against experimental data demonstrated some successes. However, accurate predictions of these tools over a wide range of conditions and channels remain elusive. Further works are required on identifying appropriate turbulent models for use in different regions. The advances in DNS demonstrated the complexity of heat transfer phenomena in simple tubes at supercritical pressures. These simulations remained focusing on the low range of Reynolds numbers, compared to those of interest to the SCWR. Extension of these simulations to large Reynolds numbers deems possible with the rapid advances in computational equipment.

Cladding temperatures in SCWR cores were analyzed at normal operations and postulated accident scenarios. The overall peaking factor of the fuel assembly on design limits has been identified as an important parameter that must be addressed properly. Applications of the limiting (conservative) or the best estimate (realistic) approach in the analyses were discussed. Analyses applying analytical codes with the same correlations and/or methodologies could introduce common errors, which are difficult to detect. Benchmarking of analytical codes for common problems is needed in support of design and safety assessments for SCWRs.

Identification of the readiness levels for various technologies is important to stakeholders and designers to move forward of the SCWR design. This would ensure appropriate levels of resources and support for R&D.

3.3. MATERIALS AND CHEMISTRY OF SUPERCRITICAL WATER COOLED REACTORS

The Technical Meeting on Materials and Chemistry for SCWRs was held at Řež, Czech Republic, 10−14 October 2016. It was the third meeting focusing on the following main objectives with emphasis on application and design issues:

- Review progress in the international initiatives and national programmes related to the R&D of materials for key components of SCWR reactors;
- Exchange recent information related to testing methodologies as well as experimental setups for examination of candidate materials for SCWR components;
- Review progress in international and national programmes related to the development of water chemistry control strategies for the SCWR;
- Support further pre-normative research and code qualification activities and their coupling.

In total, 19 participants from 9 Member States and 2 international organizations participated in the meeting.

The meeting consisted of the Opening Session, four Technical Sessions, a Topical Session, two Discussion Sessions, a Summary Session and the Closing Session. Thirteen presentations were given at the Technical and Topical Sessions, each of which was followed by very active technical discussion, and several general issues related to the session topic were discussed at the end of each Technical Session.

Based on the presentations and discussions at the Technical Sessions, the current status of R&D and remaining challenges were discussed in the area of materials and chemistry for SCWRs at one of the Discussion Sessions, and future collaborative activities were proposed and discussed at the other Discussion Session.

3.3.1. Materials selection for SCWRs

Four presentations were given in this session regarding the selection of candidate materials for SCWRs in particular for the European and Canadian SCWR concepts.

(a) Materials R&D for SCWR in Canada
Wenyue Zheng, NRCan, Canada

SCWR is one of the GEN IV reactor systems being developed through an international treaty level collaboration led by GEN IV International Forum (GIF). Following the release of the core concepts from the Japanese and the EU members, Canada published its own pressure tube based conceptual design in 2015. With an outlet temperature of 625 °C and a core pressure of 25 MPa, the Canadian SCWR concept requires cladding materials that can sustain extremely harsh in-core physical and chemical conditions. Based on initial calculations using stainless steels and nickel alloys, the maximum cladding surface temperature was predicted to be as high as 825 °C; the irradiation dose can reach as much as 10 dpa and the supercritical coolant can be very oxidizing due to the production of oxygen and hydrogen peroxide by radiolysis of light water. Under these conditions, the cladding can be readily degraded by corrosion, stress-corrosion, creep, or any of the radiation related processes such as void swelling and embrittlement. In the course of the Canadian programme (2007−2015) on R&D of in-core materials, unique experimental and computational facilities were set up specifically to probe into the behaviors of candidate alloys under these extreme conditions. Some surprising results on corrosion and SCC have been achieved and the new insights will be valuable in guiding the design and development of new alloys. The key highlights of this collaborative R&D effort were presented in this paper and the challenges for the future are also discussed.

(b) SCWR materials research at VTT
Sami Penttilä, VTT, Finland

SCWR has been selected as one of the candidate concepts for the new generation (GEN IV) of nuclear reactors. Other than the design concept itself, the choice of construction materials is possibly the most challenging technical issue. As an evolutionary step from existing LWRs, it follows the development path of modern coal fired power plants towards supercritical pressures and steam temperatures of up to 650 °C. The objectives of the work were to assess the performance of potential candidate materials under the SCW environment in terms of SCC susceptibility, creep and oxidation resistance. Based on results, the main application for ferritic/martensitic (F/M) steels will be the RPV and ex-core components, like piping where temperature is low enough (≤ 400 °C). The austenitic stainless steels were studied for the internals and fuel cladding. oxide dispersion strengthened (ODS) steels may be an alternative to replace austenitic steels at high operating temperatures but some key challenges in manufacturing processes needs to be overcome. It is clear that it is not cost efficient to select the most alloyed material, e.g. Ni based alloys, therefore, different techniques have been studied also to improve the oxidation resistance of traditional low alloyed austenitic steels, e.g. coatings and cold working. Ni based alloys are also problematic to use as core components, since their high Ni content negatively affects core neutronics. In order to estimate oxidation rates, reliable autoclave data have to be obtained and models need to be developed and validated. For this purpose, a combination of ex-situ analytical studies of the oxide film forming processes with modelling approaches was proposed. An essential part of this work was also to develop testing devices and monitoring tools capable of working at SCW conditions. This presentation highlights the key findings in materials research performed in SCW during last 6 years at VTT. Most of the work has been performed in the following projects: EU HPLWR (High Performance Light Water Reactor), EU GETMAT (GEN IV and Transmutation Materials), EU SCWR FQT and Academy of Finland projects NETNUC (new type of nuclear reactors) and IDEA (interactive modelling of fuel cladding degradation mechanisms).

(c) Summary on SCWR Materials Research Activities Performed in EC-JRC
Radek Novotny, EC-JRC

This work is a summary of experimental activities on SCWR materials research performed in EC-JRC in 2014−2016. The common objective is the selection of fuel cladding and internals materials including characterization of general corrosion resistance of pre-selected austenitic stainless, characterization of sensors, characterization of SCC resistance of pre-selected materials and development of tools for future qualification testing. In the first, tests of two high temperature electrochemical potential measuring YSZ Fe/Fe_3O_4 sensors which were installed in the SCW autoclave are introduced. Electrochemical potential measurements were performed up to 650 °C SCW. The temperature stability, the dissolved oxygen content sensitivity and long term stability at maximum temperature were investigated. The obtained data were compared to calculated values provided by the manufacturer OECD Halden Research Reactor. The sensor which performed better in the first experiments was selected for follow up feasibility study of electrochemical impedance spectroscopy (EIS) measurements in subcritical and supercritical water. Experimental setup consisted of YSZ Fe/Fe_3O_4 as a reference electrode, Pt basket as a counter electrode and 316L cylinder as a working electrode. The objectives were sequential impedance measurements to investigate effect of temperature, in particular critical point transition (from 230 °C to 500 °C with short increase to 604 °C); effect of time at selected temperature (more than 3000 h at 500 °C SCW) and effect of pressure (pressure decreases from 25−15 MPa). In-situ impedance measurements were found feasible in SCW. EIS data quality depended on SCW density and electrode setup (three electrode lower noise, two electrode setup high noise at $T > T_{crit}$). The measured impedance spectra showed combined response of SCW, SCW/steel interface phenomena and charge transfer phenomena. It was confirmed that polarization resistance: $R_p \approx 1/$ corr. rate i.e. a measure of instant corrosion rate so the sensor is suitable for long term corrosion monitoring at here selected temperatures.

In the second part, various methods how to investigate SCC susceptibility of selected materials were introduced. Most of the SCC susceptibility tests performed so far were constant extension rate tests (CERT). The work showed several examples and proposed methodology for future possible cooperative research projects either within IAEA CRP or Gif Round Robin exercise. The results of joint JRC&CANMET cooperative project on SCC susceptibility of Alloy 800H and SS 310S in 625°C 25 MPa SCW were shortly described. JRC performed three different tests (elongation rate 6.7×10^{-7} s^{-1}) with two changing parameters: water chemistry (<5 ppb, 200 ppb and 8000 ppb) and end of test criteria (elongation reached 35% and 5%). Strain rate is most probably more important than the effect of environment. It was proposed to use the similar procedure as that in NUGENIA+ MICRIN/MICRIN+ project where flat or cylindrical tapered specimens were selected to determine threshold stress for SCC initiation in light water reactor conditions.

(d) The Mutual Influence of Materials and Thermal-hydraulics on the Design of SCWR−Review of the Results of the Project "HPLWR Phase 2"
Joerg Starflinger, University of Stuttgart, Germany

Core design of SCWR is strongly influenced by the variation of thermodynamic properties of supercritical water near the critical point. Non-linearity of the property changes with temperature provides a special challenge in a hot channel analysis.

Applying the hot channel factor analysis to the expected heat up of a SCWR core resulted in the three-pass core design as analysed in the HPLWR Phase 2 project. The three passes resulted from necessity to mix the supercritical fluid during the heat up inside the core to avoid hot streaks, which could challenge the cladding material. For the HPLWR project, a maximum cladding surface temperature of 630 °C was selected. Three pass simply means that the flow passes the reactor core three times: upward flow in the core mixing in an upper plenum downward flow in a different section of the core mixing in a lower plenum upward flow in a different section and delivery of supercritical water of 500 °C to the turbine.

In the HPLWR Phase 2 project, wire wrap spacers were foreseen instead of grid spacers that are used in BWR and PWR. Such wire wrap spacers mix quite well, but could possibly lead to hot spots on the cladding surface, which must be avoided though a suitable design.

Although the flow is well mixed within an assembly, the non-uniform neutron flux (and power) profile across an assembly still results in a non-uniform heat up of the supercritical fluid. In the HPLWR design, nine smaller assemblies were grouped into one assembly cluster with common head and foot piece.

Materials and thermal hydraulics have a strong mutual interaction on design of a SCWR core. For example, thin walled components, especially fuel claddings, have corrosion problems at above 600 °C while thick walled components have no major structural problems with respect to corrosion because fossil fuel plant technology can be applied. In the design of a SCWR core, closer collaboration may be established.

3.3.2. Irradiation effects on materials and chemistry for SCWRs

Two presentations were given in this session regarding the Canadian and the Chinese SCWRs development programmes.

(a) State of the Art in Cladding and Fuel Channel Material Selection for Canadian SCWR Concept
Lori Walters, Canadian Nuclear Laboratories (CNL), Canada

The Canadian SCWR concept requires materials to operate at higher temperatures than current GEN III water cooled reactors. Materials performance after radiation damage is an important design consideration. Materials that are both corrosion resistant and radiation damage tolerant are required. Although the extreme conditions and the broad range of SCWR in core operating conditions present significant materials selection challenges, candidate alloys that can meet the performance requirements under most in-core conditions have been identified. However, for all candidate materials, insufficient data are available to unequivocally ensure acceptable performance. This presentation summarized the knowledge gaps regarding the performance of candidate in-core materials and suggests experiments and data needed to verify their viability. Research programmes are to include out of pile tests on unirradiated and irradiated alloys, together with in-pile tests in a SCW loop to be constructed.

(b) Conceptual Design for SCWR Material Irradiation Test in NPIC
Wang Hai, Nuclear Power Institute of China (NPIC), China

In order to choose appropriate materials for internal structures and fuel claddings of SCWR, it is necessary to perform in-pile irradiation tests under high temperature environment to check the characteristics of the candidate materials that will be selected with out of pile test results.

A conceptual design of a material irradiation test is ongoing for candidate materials for internal structure and fuel claddings of the CSR1000 concept developed by NPIC. The test will be performed at the test reactor called HFETR in NPIC, which has more than 30 year experience in operation and post irradiation examination (PIE). It is planned to design and manufacture the test rig and online measuring system in 2017, to conduct in-pile irradiation test for 1−4 kinds of prospective candidates in 2018, and to perform the PIE and comprehensive evaluation in 2019. The PIE will include thermophysical property measurement and mechanical property testing, such as impact, tensile, toughness and so on. The irradiation data and PIE results is expected to be an essential reference for CSR1000 design. An FQT for a small scale CSR1000 fuel assembly with a research reactor of NPIC is also being planned. Engineering design of the test loop and rig for the functional qualification test will be started in 2019. The loop will simulate the actual operating condition to qualify the fuel assembly of CSR1000 and to check the characteristics of the fuel assembly.

3.3.3. General corrosion and stress corrosion cracking (SCC)

Five presentations were delivered to summarize recent progress on general corrosion and SCC of candidate materials in Canada, China, Romania and Spain, and the possibilities to mitigate corrosion problems were discussed.

(a) Effect of High Temperature Steam Exposure on Oxidation Behaviour of NiCrAl and FeCrAlY
Xiao Huang, Carleton University, Canada

In this study the FeCrAlY and NiCrAl coating samples were tested in steam at 800 °C for 300 and 600 hours. The FeCrAlY became discolored rapidly, while the NiCrAl still maintained some metallic sheen after 600 hours. The weight change results suggest that more oxide formation took place on FeCrAlY than on NiCrAl. In particular, grain boundary oxide (Al_2O_3) formed on FeCrAlY surface upon exposure to steam after 300 hours. Further exposure caused more intragranular Al_2O_3 to form, in addition to magnetite formations on the grain boundary regions. For the NiCrAl samples, NiO formed after steam exposure for 300 hours. Spinel and $(Cr,Al)_2O_3$ were also found after 300 hours, along with very limited amounts of Al_2O_3. After 600 hours in steam, Al_2O_3 became well developed on NiCrAl, and the coverage of spinel and Cr_2O_3 on the surface was reduced.

(b) SCWR Candidate Fuel Cladding Materials and Their Corrosion Behaviours
Lefu Zhang, SJTU, China

On current available data, unsatisfied general corrosion and stress corrosion cracking (SCC) properties still remain the major problems among the candidate materials proposed for making fuel cladding of a supercritical water cooled reactor (SCWR), mainly due to the high temperature and corrosive water environment. Ferritic/martensitic (F/M) steels, austenitic stainless steels, nickel base alloys, alumina forming steels, and their oxide dispersion strengthened (ODS) materials have been studied during the past years. Austenitic stainless steels, type 310S, 316L/316Ti and 800H, have been short listed and studied intensively. More recent results confirmed their susceptibility to SCC at SCWR operating temperatures. Scientists are looking for better candidates for fuel cladding and trying to innovate new materials with improved strength and lowered corrosion rate in high temperature and pressure water environment. ODS is an effective method in promoting the strength of base metal while reducing its susceptibility to SCC. The F/M and austenitic stainless steel based ODS steels have been prepared and studied by several institutions, showing promising properties at both mechanical and chemical aspects. However, intensive study is still needed for characterization of fuel cladding material for our test SCWR. Future tests are defined for characterizing and verifying the reliability of the short listed materials, and general corrosion and SCC tests are still the major research works under plan.

(c) Overview of Materials for SCWR Tested in RATEN ICN
E.M. Fulger, RATEN ICN, Romania

The most promising structural materials for the SCWR are austenitic stainless steels and nickel base alloys. Literature review showed that the most probable fuel cladding material may have an austenitic structure and contain high Cr concentration up to 22% or higher. Therefore, a systematic study on the corrosion behavior of structural materials is needed to ensure their safe application to nuclear reactor systems.

This paper presented a part of the research performed at RATEN ICN on oxidation in supercritical conditions of commercially available austenitic alloys. The work was focused on investigating the oxidation behaviour, surface morphology and microstructure of some candidate materials: austenitic stainless steels (304L, 310S, 316L, 321) and Ni based alloys (alloy 800HT and alloy 718) in water at two different temperatures (550 °C ; 600 °C) and 25 MPa pressure. After exposure in supercritical conditions, the studied alloys were characterized using various methods: gravimetry, optical microscopy, scanning electron microscopy (SEM), energy dispersive spectroscopy (EDS), and X ray diffraction (XRD).

A comparison of the alloys provides interesting information. Oxidation was observed to be the main form of high temperature water corrosion experienced by the tested alloys. In terms of mass gain, the 310S steel had the best behavior comparatively with the other samples. The weight gain in supercritical water decreased with increasing Cr content, in order: alloy 316L > 321> 304L > 718 alloy > 800HT > 310S. Generally, the austenitic alloys developed dual layered oxide scale in supercritical water, consisting of Fe_3O_4 for the outer layer and spinels (Ni, Fe) Cr_2O_4 for the inner layer. Both Ni base alloys, Inconel 718 and Incoloy 800HT, developed a dual layered oxide consisting of a nickel /iron rich outer layer and a chromium rich inner layer. In case of Inconel 718 complementary to oxidation, pitting corrosion has been observed but in time, oxidation (which is more predictable) was dominant.

(d) Effect of Intergranular Carbides in the Oxidation and Stress Corrosion Behavior of Alloy 690 in Supercritical Water
A. Sáez-Maderuelo, CIEMAT, Spain

The nickel base alloy 690, which was designed as a replacement for the nickel base Alloy 600, is a material widely used in the nuclear industry due to its optimum behavior to SCC under nuclear reactor operating conditions. Because of this superior resistance to some degenerative processes, the Alloy 690 has been proposed as a candidate structural material for the SCWR, which is one of the designs of the next generation of nuclear power plants (GEN IV).

It was expected that, as in the Alloy 600 TT, the presence of intergranular carbides in the Alloy 690 could improve its resistance to SCC in LWR environments. However, some authors have observed higher resistance to SCC in specimens without intergranular carbides than in specimens with intergranular carbides [38], [66]. Therefore, the role of the intergranular carbides in the Alloy 690 is still an open question.

Considering these results, the aim of this work is to study the oxidation behavior of Alloy 690, as a preliminary step to SCC, with and without intergranular carbides in deaerated SCW at 25 MPa and at 400 ºC and 500 ºC in order to gain some insight into the understanding of the effect of these carbides in the resistance of this material. Results from this work show an optimum oxidation behavior of Alloy 690 thermally treated and solution annealed in deaerated SCW at both temperatures. However, the absence of intergranular carbides seems to promote the oxidation processes on Alloy 690.

(e) The effect of dissolved oxygen on stress corrosion cracking of 310SS in SCW
Jinhua Liu, Nuclear Power Institute of China (NPIC), China

Austenitic stainless steels have been widely used as the major structural materials due to excellent combination of mechanical properties and corrosion resistance in high temperature. 310 SS has been regarded as a most promising material of fuel cladding for SCWR. It is well known that SCC of austenitic stainless steels is influenced by water chemistry, such as oxygen. In order to obtain a deep understanding of the SCC behavior of 310 SS in SCW, more research is urgently needed.

The effects of dissolved oxygen content on the tensile properties and SCC susceptibility of austenitic stainless steel 310SS were studied by performing slow strain rate tensile (SSRT) tests. The SSRT tests were carried out in supercritical water at temperature of 620 ℃, a pressure of 25 MPa, and a strain rate of 7.5×10^{-7} s^{-1}. The dissolved oxygen content was included 0, 500, 1000, 2000, and 8000 ppb. The SSRT tests were performed in a supercritical environment corrosion testing machine system, which can control the DO concentration, pH and conductivity.

After the tests, strain–stress curves were analyzed to identify the mechanical properties of 310SS. The morphologies of the side surface and fracture surface on the specimens were conducted by scanning electron microscope (SEM), and the chemical composition and structure of the oxide formed were examined by energy dispersive spectroscopy (EDS), in order to investigate fracture mode and to evaluate SCC susceptibility.

The results show that the elongation decreased dramatically with the increasing of dissolved oxygen concentration. Cracks on 310SS were widely distributed over the whole gauge section and a brittle fracture mode was observed on the fracture surface. The Cr content in the oxide layer on the surface showed significant increase with the increasing of DO concentration. The corrosion products formed on 310SS were Fe/Cr oxide layers, and they are classified into two layers, an outer Fe rich oxide layer and an inner Cr rich oxide layer.

3.3.4. Improvement and design of materials for SCWRs

One presentation was given in this session on chemical composition modification of austenitic stainless steel type 310S.

(a) Chemical composition modification of 310 type stainless steel
Qiang Zhang, Nuclear Power Institute of China (NPIC), China

Austenitic stainless steel (ASSs) 310SS is a promising fuel cladding material for SCWR, due to high strength, corrosion resistance, oxidation resistance, low sensitivity to neutron irradiation and good processing ability. However, a large amount of coarse $Cr_{23}C_6$ particles would be precipitated on the grain boundaries of 310SS, which results in the increase of brittleness and the deterioration of corrosion resistance. Minor additions of strong carbide forming elements, such as Nb, Ti, Ta, Zr, could suppress the formation of $Cr_{23}C_6$.

On the basis of 310SS (25Cr-20Ni-0.08C wt.%), the cluster plus glue atom model is introduced to design new alloy by multi-element co-alloying (Nb, Ti, Zr, Ta, W) of 310S. The purpose is to improve its mechanical properties and microstructure stability.

This cluster model dissociates the solid solution structure into a cluster part and a glue atom part: the cluster is the nearest neighbor polyhedron and glue atoms are located between the clusters. It is found that the stable FCC solid solutions generally correspond to the cluster formula of [CN12 cluster](glue atoms)$_{1\sim6}$, where the cluster is a cub octahedron with a coordination number of 12. The basic Fe–Ni–Cr ternary composition of 310SS ($Fe_{55.0}Cr_{24.7}Ni_{22.3}$ wt.%) is determined as the cluster formula [Cr–

$(Fe_{10}Ni_2)](Cr_4Ni_2)$, where Cr represents Cr similar elements (Cr,Nb,Ta,Ti,Zr) and Ni represents Ni similar ones (Ni,Mn,C).

A new alloy Fe-24.6Cr-22.2Ni-1.01Mo-0.09Nb-0.09Ti-0.17Ta-0.05C is designed from this formula by Nb, Ti and Ta co-alloying. The alloy ingots were prepared by vacuum arc melting processing. These ingots were then solid solutioned at 1200 °C for 1h, stabilized at 950°C for 0.5 h, and aged at 800 °C for 24 h. The experimental results indicate that after stabilization treatment, a large amount of (Nb,Ta)C nanoparticles with a size of 50−70 nm are distributed on the grain boundaries of the matrix, besides minor TiC and $Cr_{23}C_6$. After aging treatment, the MC nanoparticles dispersed in the inner grains uniformly, with no change of the particle size; a few $Cr_{23}C_6$ particles precipitate on grain boundaries, with a size of about 1 μm. It is suggested that the addition of Nb, Ti and Ta can form fine TiC, (Nb,Ta)C particles and the coarse $Cr_{23}C_6$ particles are suppressed.

3.3.5. General discussion on materials and chemistry for SCWRs

Recent studies have shown the possibility of changes in the oxidation rate of materials with pressure and temperature. The variation of these parameters could change the behaviour of water from gas like fluid to liquid like fluid or reversely. These changes could affect the corrosion resistance of candidate materials. More tests are necessary in order to gain an in depth knowledge.

Despite of the development of advanced fuel assembly (AFA) and ODS steels for other GEN IV concepts (such as lead cooled fast reactor or very high temperature reactor), there is no material currently available that would demonstrate sufficiently low corrosion resistance at peak cladding temperatures above 600 °C. Thermomechanical processing (TMP) has been proposed to increase general corrosion resistance of materials and at the same time maintain high stress corrosion cracking resistance.

Two promising surface treatment technologies to increase general corrosion resistance of stainless steels were introduced: coatings and surface finish. The beneficial effect of shot peening was demonstrated in many projects on development of material for supercritical coaled power plants. Convincing results of corrosion tests of shot peened stainless steels in SCW. Discussion revealed that similar studies have been under way in other organizations and they are subjects of other European and international projects. However, it is unclear of the effect on the microstructure of a shot peened cold worked layer under the exposure to high temperature above 600 °C within the foreseen cladding service life. Furthermore, stress and strain on the materials could have an adverse impact on coatings for their long term corrosion resistance. If the microstructural of the bulk material changes, the integrity between the substrate and the coating could be lost. Promising results were presented on CrN coatings deposited by PVD on 316L substrate in Halden Research Reactor. However, further studies are needed to verify the SCC resistance.

An electrochemical potential (ECP) sensor is required for future in-pile test in CVR in order to be able in the future to verify the effect of radiolysis on water chemistry in SCWR environment and help to determine concentrations of the most important constituents such as dissolved O_2 content, H_2 and H_2O_2. Applications of ECP and electrochemical impedance spectroscopy measurements using YSZ Fe/Fe_3O_4 pseudo-reference electrode were demonstrated for temperatures up to 650 °C. The issue of reliability of OCP sensor remains. The calibration of reference electrode for future radiolysis studies performed in in-pile CVR facility as well as the design of such sensor remain to be challenges.

The assessment technique was examined for SCC susceptibility of selected materials. Current SCC SSRT tests primarily used for screening purposes might be applicable for assessing SCC susceptibility. However, more sophisticated tests are needed including fracture mechanics crack growth rate SCC tests as well as SCC initiation tests. In future round robin exercise on SCC susceptibility organized within GIF SCWR PMB, the procedure which was successfully used in NUGENIA+ project MICRIN/MICRIN+, where tapered tensile specimens were used to determine stress threshold for SCC initiation assessment of model duplex steel and alloy 182, could be applied. Up until now only few crack growth rate tests in SCC were performed in JRC and SJTU. There are limits to use the elastic fracture mechanics approach. Further development of procedure is required to perform crack growth rate tests.

Among cross cutting activities between thermal hydraulic, materials and chemistry, collaborations among experts in these technology areas are required, such as the examination of the effect of surface roughness on character of flow or effect of oxide layer growth on heat transfer efficiency of the fuel cladding.

The number of studies assessing the effect of irradiation on corrosion and SCC was limited. The majority of these studies were performed using electron, proton or heavy ion irradiated materials. An in-pile facility is essential to make further progress in the above mention areas as well as in the radiolysis research. In-situ irradiation assisted stress corrosion cracking tests still presents the biggest challenge, although a test facility has been proposed for in-situ irradiation in the research reactor at NPIC. Most data on irradiation effects on mechanical properties for selected materials and relevant temperatures are either available or can be obtained via dry irradiation in dedicated channels at research reactors.

4. CONCLUDING REMARKS

The SCWR is an innovative WCR concept that uses supercritical pressure water as reactor coolant. The use of supercritical pressure water as coolant enables achieving higher thermal efficiency and implementing simpler heat transport system configurations than those of conventional WCRs. Hence, the SCWR concepts have the potential for improved economics and sustainability.

The concept of the SCWR dates back to 1950s; the US proposed a few SCWR concepts. Significant R&D activities on concept development started in 1990 in Japan. Several Member States started their activities after 2002, when the SCWR was selected as one of the next generation nuclear energy systems. Since then, these Member States have been engaged in R&D activities to develop their own concepts and to advance the associated technologies under national programmes and international collaboration.

A few concepts, i.e., Canadian SCWR, HPLWR and JSCWR, have been developed and assessed as technically feasible so far. Several other concepts are under development. The design of SCWRs is very flexible. For the reactor type, the Canadian SCWR is pressure tube type with thorium/uranium fuel cycle; and the others are pressure vessel type with uranium fuel cycle. Regarding the neutron spectrum, some have thermal spectrum, and others have fast spectrum.

4.1. FIRST STEP OF DEVELOPMENT

The first step of the development is to improve understanding of key phenomena that are specific to conditions and environments encountered under SCWR operations. At this step, research institutes and universities have been playing a major role in R&D, and several reactor vendors have also been involved.

Three major R&D activities have been focusing at research institutes in support of the SCWR development: thermal hydraulics, materials and chemistry. The state of the art R&D is summarized below for each area:

- Small scale bundle and simple geometry tests carried out in many institutes provided relevant thermal hydraulics data to enhance understanding of thermal hydraulics phenomena and validate thermal hydraulics technology of interest to the design of SCWRs;
- Material testing at various laboratories enhanced understanding of degradation mechanisms under supercritical water conditions and generated a database for identifying appropriate material candidates for in-core and out of core components in SCWRs;
- A strategy for reactor coolant chemistry has been formulated from vast R&D and operational experiences of BWRs and supercritical fossil fuel power plants.

So far, understanding of key phenomena and advancement of technologies have been achieved in various Member States. Fundamental research has been completed successfully to the level to understand the important phenomena and to provide information necessary for the next step.

4.2. SECOND STEP OF DEVELOPMENT

The second step in R&D is to support the advancement of engineering and design of SCWRs including assessment of performance and safety. At this step, industry and regulatory body involvement may be enhanced, and availability of suitable infrastructures as well as continuous training of highly qualified personnel may be assured. Much closer international collaboration may be beneficial.

Two significant projects have been identified. One of them involves in-pile fuel tests to confirm thermal hydraulics characteristics and material selection for proposed water chemistry strategy. The designs of test facility and test section have been completed. The other project includes integral tests to demonstrate safety. These projects are anticipated to take around ten years for completion.

4.3. THIRD STEP OF DEVELOPMENT

The third step is to design and construct a SCWR prototype and/or demonstration reactor in order to confirm its performance, operability and economics. The SCWR is an innovative concept, proposed as a natural evolution of the most advanced WCRs and supercritical fossil fuel power plants.

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ABBREVIATIONS

ADS	Automatic Depressurization Systems
AFS	Auxiliary Feedwater Systems
ASS	Austenitic Stainless Steel
AFA	Advanced Fuel Assembly
ATF	Accident Tolerant Fuel
BOP	Balance of Plant
BWR	Boiling Water Reactor
CERT	Constant Extension Rate Tests
CFD	Computer Fluid Dynamics
CRP	Coordinated Research Project
CSR1000	Chinese SCWR concept
CVD	Chemical Vapor Deposition
DNS	Direct Numerical Simulation
EAC	Environmental Assisted Cracking
ECP	Electrochemical Potential
ECCS	Emergency Core Cooling Systems
EDS	Energy Dispersive Spectroscopy
FA	Fuel Assembly
FFSCWPs	Fossil Fuel Supercritical Water Plants
F/M	Ferritic/Martensitic (steel)
FQT	Fuel Qualification Test
GEN IV	Generation IV reactors
GIF	GEN IV International Forum
HEM	Homogeneous Equilibrium Model
HP	High Pressure
HPLWR	High Performance Light Water Reactor
HTC	Heat Transfer Coefficient

HTD	Heat Transfer Deterioration
HTE	Heat Transfer Enhancement
HWR	Heavy Water Reactor
IASCC	Irradiation Assisted Stress Corrosion Cracking
IP	Intermediate Pressure
ISSCWR	International Symposium on SCWRs
JSCWR	Japanese SCWR concept
LEU	Low Enriched Uranium
LOCA	Loss of Coolant Accident
LP	Low Pressure
LPCIs	Low Pressure Coolant Injection system
LWR	Light Water Reactor
NPP	Nuclear Power Plant
ODS	Oxide Dispersion Strengthened (steels)
PIE	Post Irradiation Examination
PT	Pressure Tube
PV	Pressure Vessel
PVD	Physical Vapor Deposition
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
PCT	Peak Cladding Temperature
RHR	Residual Heat Removal
RP	Reactor Plant
RPV	Reactor Pressure Vessel
SCC	Stress Corrosion Cracking
SCW	Supercritical Water
SCWRs	Supercritical Water cooled Reactors
SCP	Supercritical Pressure
SEM	Scanning Electron Microscopy

SSRT	Slow Strain Rate Tensile
TECDOC	Technical Document
TMP	Thermomechanical Processing
TRU	Transuranics
VVER-SCP	Russian SCWR concept
WCRs	Water Cooled Reactors
XRD	X Ray Diffraction

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Technical Meetings

Heat Transfer, Thermal Hydraulics and System Design for SCWRs, Sheffield, United Kingdom of Great Britain and Northern Ireland: 22–24 August 2016

Materials and Chemistry for SCWRs, Řež, Czech Republic: 10–14 October 2016

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19-00251

International Atomic Energy Agency
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ISBN 978–92–0–101919–6
ISSN 1011–4289