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Review

Design concepts of supercritical water-cooled reactor (SCWR) and nuclear marine vessel: A review

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ABSTRACT

Supercritical Water-Cooled Reactors (SCWRs) conception has been developing as promising advanced nuclear systems in recent decades around the globe because of its many desirable features. It has many potential advantages include high thermal efficiency (~45% in comparison to ~33% efficiency of present Light Water Reactors), relative simplicity in plant construction with high power density, which eventually would decrease the investment expenses so that the reactor economy will be improved. SCWR is the only direct upgrade version of the Generation III water-cooled reactor using water as a coolant material amongst the six reactor types being studied by the GEN-IV International Forum (GIF). Generally, SCWR design is broadly categorized as (i) pressure-vessel type and (ii) pressure-tube type. Japan first suggested the pressure-vessel type concept and then by the Euratom partnership. The pressure tube idea was proposed by Canada first, and therefore it is named Canadian SCWR. These two concepts have several similar characteristics; outer pressure and temperatures, selection of steam cycle, features of materials, and heat transfer. In the present work, a critical analysis of existing SCWR conceptual designs was performed. An assessment of these land-based designs and a review of current models of nuclear submarine and ship were conducted. Also, considering the operating conditions and performance requirements of performance for ships, an optimal design of SCWR for ships was proposed. So far, pressurized water-saturated steam nuclear power plants have been demonstrated to be reliable marine power plants. Further technology and design improvements have been made to make LWR plants competitive for a large number of marine applications. The conceptual design of SCWR for ship application, which is preliminarily discussed in this paper, could represent potential progress in marine life technology. This advancement is supposed to result from a significant reduction in plant size, weight, and hopefully, capital and operating costs. The confirmation of this design necessitates extensive analysis and experiments.

1. Introduction

Nuclear power is a sustainable, economical, eco-friendly, and promising solution to the world's energy crisis. At present, operating nuclear power reactors are mostly Generation II type and are contributing 14% of the world's generation of electricity. The new generation of nuclear power plants supposed to have a significant improvement in safety and economy, minimal waste generation, and proliferation-resistant, which are the key considerations in future energy systems. To achieve the ambitious goal, The GEN-IV International Forum (GIF), a collaborative international organization, has been carrying out the necessary R&D to improve the feasibility and performance of the generation IV nuclear systems. Supercritical Water-Cooled Reactors (SCWRs) (Oka and Koshizuka, 1993) are promising future nuclear energy systems due to their high thermal efficiency and their comparative plant construction simplicity. SCWR, as shown in Fig. 1, is the only generation IV nuclear reactor using water as coolant material. It is a combination of the traditional Light Water Reactor (LWR) and the

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Supercritical Fossil Power Plant (FPP) and will maintain 25 MPa core pressure and 500 °C average core exit pressure. Due to the high exit temperature of the coolant of SCWR, the thermal efficiency reaches about 45%, which is step up more than 50% compared with conventional LWRs. Even though various researches had been carried out for power generation using SCWR, several issues and challenges stay behind unsettled. In this work, the design features (core and fuel assembly designs) are analyzed by collecting existing literature on the design of water-cooled reactors in supercritical water conditions.

In this work, conceptual design analysis of existing SCWR was performed qualitatively. A comparative assessment of different SCWR designs and a review of reactor designs for submarine and ship were conducted. The SCWRs will have elevated economic efficiency with no limitation of the boiling temperature, with a new compact structure devoid of the recirculation system, steam separator, and dryers of BWRs, which is appropriate for transportation applications. Hence, the new generation reactors designs for power plant would be downscaled for the utilization in the ships. In this work, the design features (core and fuel assembly designs) are analyzed by collecting existing literature on the design of water-cooled reactors in supercritical water conditions. A state-of-the-art review is implemented. Besides, some models with features suitable for nuclear ships are highlighted. A preliminary overall conceptual nuclear reactor design for nuclear ship application is also discussed with some features ideal for nuclear ships are highlighted. Moreover, the present study deals with the SCWR technology review and evaluation for the application in the nuclear-powered ships. In SCWR, using water as a coolant material is one of the best choices due to its high heat capacity and thermal conductivity. It is enormously available in the ship's operating environment. Further, notably, the highly compact SCWR system is appropriate for transportation applications.

Therefore, the supercritical plant could be a step-change in the marine industry. Considerable progress in plant performance would be likely by the supercritical steam conditions. An additional degree of design flexibility would be attained, which, when fully utilized, promises significant economic rewards.

To accomplish the stated tasks, concentrated has been made on the study of several specific problems as follows:

- Analysis of existing land-based SCWR: The SCWR has been chosen by GIF to analyze as a promising next-generation reactor design concepts. The steam cycle of SCWR reactors will be derived from advanced FPP, but the reactor core, and in the fuel assembly is still needed to be developed. Moreover, no works have been performed on SCWR for ship applications. Therefore, a systematic review of existing SCWR reactor core and fuel assembly design concepts have been analyzed in addition to the overall designs of SCWR.
- Innovative designs of SCWR components: Some innovative designs, especially the core part of SCWR components, are reviewed and summed-up.
- Nuclear submarine reactor characteristics: The submarine reactor design review is one of the targets of this study.
- Special needs of nuclear reactors for nuclear ships: Based on the necessities, such as high power volume density, which gives the nuclear ship a higher speed and a substantial energy capacity for an unlimited navigation range, etc. Some distinctive evaluations of the proposed nuclear reactor designs are made.

2. Working principle and design concepts of SCWR

SCWRs are water-cooled reactors having high-temperature (374 °C or 705°F), high-pressure (22.1 MPa or 3208 psia), and operate above the thermodynamic critical point of water (DoE, 2002). At critical point, water does not behave either a liquid or a gas but instead reaches a distinct state withholding unique characteristics. The SCWR is a combination of the traditional Light Water Reactor (LWR) and the supercritical Fossil Power Plant (FPP). The reactor will maintain a core pressure of 25 MPa and an average core exit temperature of 500 °C. As the coolant ditches the reactor core, the high coolant temperature offers the SCWR the possibility for high thermal efficiency (45%), which is step up more than 50% compared with conventional light water reactors.

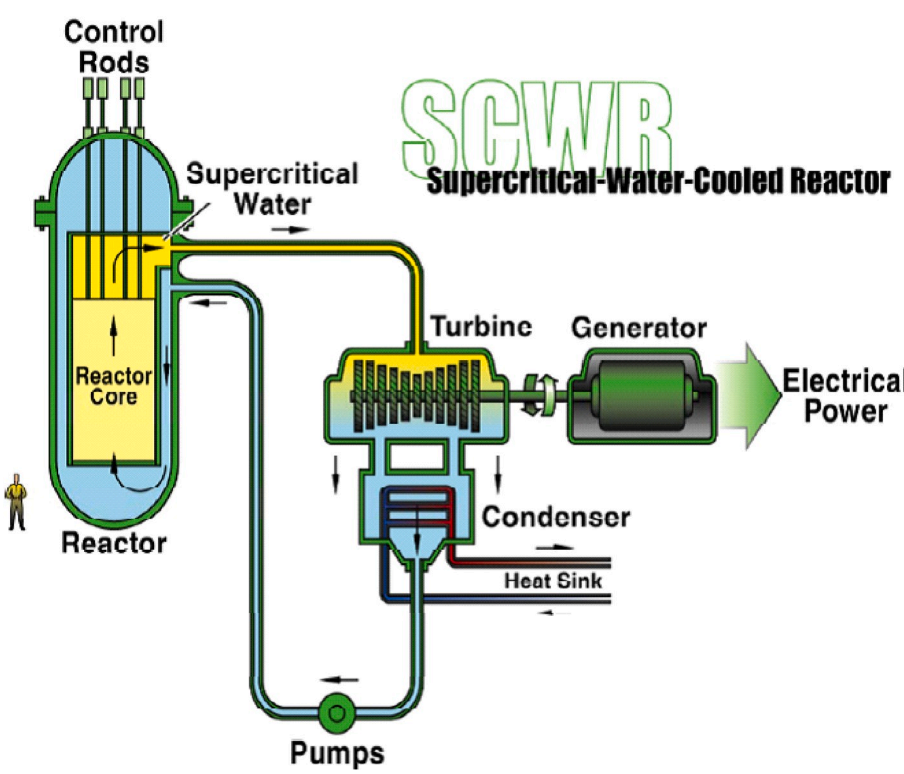

Fig. 1. A typical SCWR system.

Even though various researches had been carried out for stationary power generations using SCWR, several issues and challenges stay behind unsettled. The elimination of steam dryers, steam separators, recirculation pumps, and steam generators make SCWR a simpler plant with minimum major components. The higher enthalpy content of the coolant results in a lower-coolant mass flow rate per unit core thermal power. It reduces the size of the reactor coolant pumps, piping, and related equipment, and the pumping power. However, material degradation issues at SCWR core conditions, as well as thermal-hydraulics and physics of SCWR, require much work. The design of SCWR can be as a thermal or fast reactor cooled by supercritical water. It is considered an evolution of actual BWR (Boiling Water Reactor). Basing on different category criteria, SCWRs can be sorted into different types. In this paper, SCWRs will be sorted by two approaches, one is the neutron spectrum, and the other is reactor vessel types. Most nuclear reactors are thermal reactors, which means the radioactive fuel absorb slow neutrons to fission and producing heat in the meantime. While the fission will produce fast neutrons with very high velocity, this means moderators need to be used in the reactor pressure vessels to slow down the speed of the neutron. A breeder reactor with the ability to convert fissionable materials to fissile materials at the same time producing heat energy. Another categorized method bases on the type of reactor pressure vessel. Most nuclear fission reactors use two main types of vessels: (1) a large pressure vessel (PV) maintaining wall thickness of nearly 20 cm to accommodate heat source of the reactor core, such as the typical LWRs; (2) distributed pressure tubes (PT) used in Canadian deuterium uranium reactor (CANDU) and RMBK designed by USSR. There are 14 members in the GIF (GIF, 2017). Each member state is focusing on specific types of next-generation nuclear reactors. Among them, seven countries have already done lots of research work in developing SCWR. Table 1 shows a summary of typical SCWR overall designs in these countries.

SCWR combines technologies of verified modern nuclear and FFPs jointly with some upgrades to fulfil the design goals of GIF. These include safety enhancement, better sustainability, and economics, as well as enhanced proliferation resistance. It is a noticeable improvement because both nuclear and supercritical FPPs are proven technologies with many years of operating experience. Fig. 2 shows the merged nuclear and supercritical FPP technology for SCWR development. Established advanced concepts, like the turbine technology and systems of supercritical water, e.g., improved passive safety systems, can also be implemented as further development. However, it is crucial to integrate nuclear and fossil technologies.

The main design objective of SCWR is the economic improvement that can be attained by increasing the outlet temperature of the core — the results in higher thermodynamic efficiency in comparison with the present fleet of nuclear reactors. Depends on the design, SCWRs have thermodynamic efficiency ranges between 40 and 48% compared to 33% in the case of current reactors. Several plant design simplifications, like, direct thermodynamic cycle analogous to BWR has been considered in SCWR lead to cost-saving significantly. The flexible design of SCWR adopts thermal and fast spectrum concepts and thereby provides advanced or conventional fuel and fuel cycles. The main objective of SCWR is electricity generation, though it is suitable for desalination, hydrogen production as well as heat productions.

The conceptual designs of SCWRs generally categorized into two classes: Japan and European partnership proposed pressure vessel concepts, and Canada proposed pressure tube concepts. Besides the specifics of the core design, the two ideas have many identical characteristics, such as outlet pressure and temperatures, heat transfer characteristics, materials, steam cycle options, etc. Hence, the research & development work for these two reactor types are nearly similar, which enabled them to pursue collaborative research. In the present study, the emphasis has been made on the review of these reactor types, especially in the core and fuel assembly design concepts. The following chapters examine various developed design ideas until now to fulfill the design objective. The main focal point will be on the concepts of the thermal reactor, though besides, a fast neutron spectrum shall also be addressed.

2.1. EU HPLWR design concept

In Europe, ten partners from eight European countries have been investigated SCWR design under a collaborative research project named HPLWR. A first study has been accomplished to evaluate the concept and plant features under the 5th Framework Program, where a supercritical pressure of about 25 MPa and a coolant temperature from 280 to 500 °C have been considered (Squarer et al., 2003). The thermal power of the reactor is considered to be 2300 MW with a net electric output of 1000 MW, and a net efficiency of the steam cycle is 43.5% (Pioro, 2016). In a once-through cycle, the high-temperature steam is passed directly to the high-pressure turbine (Bittermann et al., 2005). The current three-pass core HPLWR reactor pressure vessel with internals is illustrated in Fig. 3.

2.1.1. Fuel assembly design concepts

A suitable assembly design has been considered to optimize the axial power profile with constraint density differences of coolant inside the core. A significant part of the design works has been performed by Oka and Koshizuka, The University of Tokyo, in the 1990s (Oka and Koshizuka, 1993). To flatten the axial power profile, Dobashi et al. (1998) offered the design of the first fuel assembly for a thermal neutron spectrum using extra water rods. Fuel pins arranged hexagonally and cooled by rising coolant has been used that were contained in an assembly box similar in a BWR. Inside this assembly, the required extra moderator at close to the core top was supplied through water tubes.

Feedwater was flowing, preferably downwards across these water tubes will be merged at the bottom of the core with added feed water. To minimize moderator water temperature that is caused by the hotter coolant, these water tubes were thermally insulated. For the first HPLWR concept, this assembly design is used by Squarer et al. (2003) as a reference. He opted that due to the non-homogeneous radial power distribution, various enrichment required in some fuel pins to homogenize the coolant temperature. A square fuel pin has been arranged by Yamaji et al. (2005a,b) with up to 36 extra square water tubes with the moderator at the core's upper part to solve this issue. Cheng et al. (2003) showed an analogous design using 25 water tubes of square shape and

Table 1
Summary of typical SCWR overall designs.

	Japan	EU	US	Korea	Canada	Russia	China	Japan	Russia
Reactor Layout	PV	PV	PV	PV	PT	PT	PV	PV	PT
Neutron spectrum	thermal	thermal	thermal	thermal	thermal	thermal	thermal	fast	fast
Thermal power (MW)	2740	2188	3575	3846	2540	1960	2300	3893	2800
Electrical power (MW)	1217	1000	1600	1700	1140	850	1000	1728	1200
Thermal efficiency (%)	44.4	44.0	44.8	44.0	45.0	42	43.5	44.4	43
Core pressure (MPa)	25	25	25	25	25	25	25	25	25
Inlet temperature (°C)	280	280	280	280	350	270	280	280	400
Outlet temperature (°C)	530	500	500	508	625	545	500	526	550
Moderator	H_2O	H_2O	H_2O	ZrH_2	D_2O	D_2O	H_2O	/	/
Fuel enrichment (%)	6.1	6	5	5.8	9	6	5.6	/	/

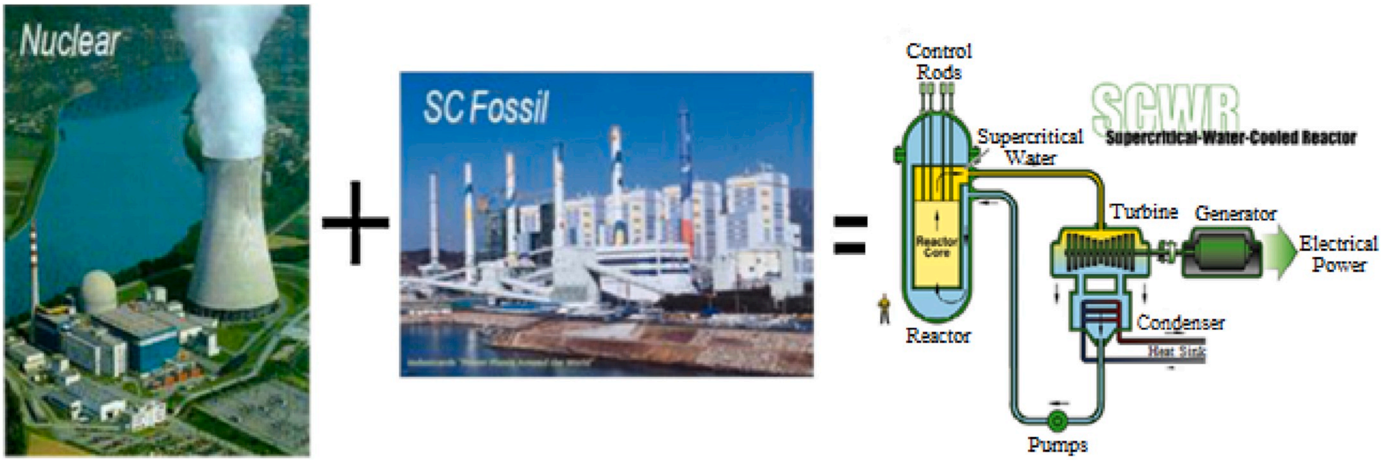

Fig. 2. Merger of nuclear and supercritical fossil plant technology for SCWR development (Courtesy of Saha et al., 2013).

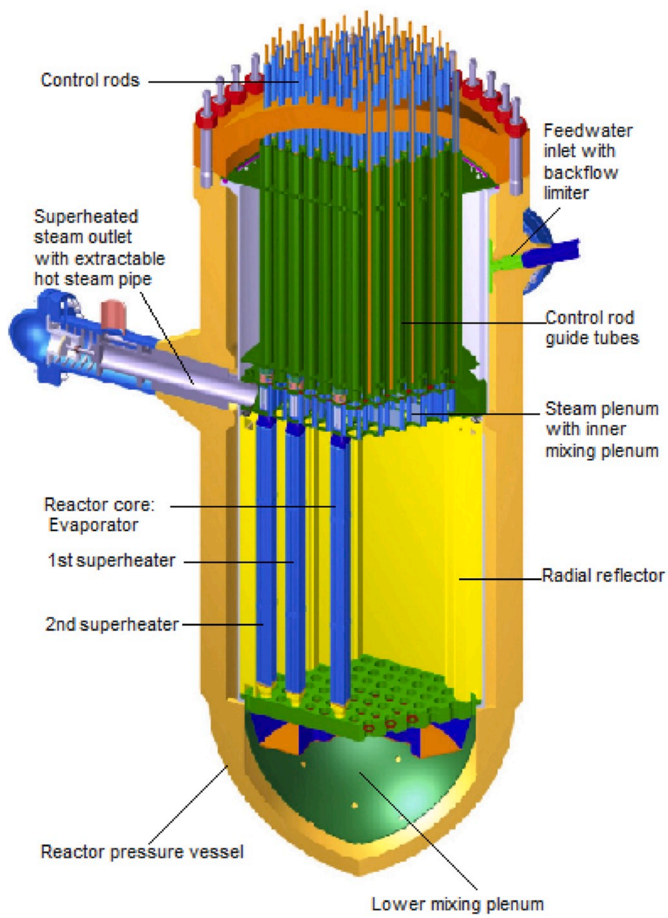

Fig. 3. Design of the assembled RPV with internals for the three pass core (Courtesy of Fischer et al., 2009).

an advanced assembly of hexagonal shape, which exemplified as a choice of HPLWR core design studies. To overcome the complexity of added inside assembly water rods', Boungiorno (2003) considered the hexagonal and smaller assembly of 19 fuel pin with every single pin was moderated by coolant water inside gaps between the fuel assembly boxes. However, the inner fuel pin will be moderated again, which indicates the necessity of higher enrichment. Joo et al. (2005) proposed higher assemblies with quite homogeneous power distribution. A cross-shaped zirconium hydride rod added to the assemblies to adjust the missing moderation of the coolant. Inside these ZrH rods, control rods were supposed to run. A design study conducted by Hofmeister et al. (2005) using a square arrangement analogous to a BWR assembly, and every single assembly comprises a water tube with 40 fuel pins. They combined and optimized these concepts with the provision that each fuel pin must be placed adjacent to any moderator water. To optimize the power density, the ratio of the moderator to fuel must be near to a PWR, and to minimize the fuel enrichment, structural material to fuel ratio should be kept a minimum. Nevertheless, control rods of cross-shaped type was inserted from the core top to the water tubes in a different design from BWR design. Such kind of small assemblies might necessitate individual control rod drives. Nine of such assemblies are grouped to a cluster, with a single head and foot part, which will also ease handling during maintenance. The clusters may be changed positions between superheater and evaporator. The outer diameter of the fuel rods is 8 mm, having fuel pellets with 4.2 m active core height. To improve mixing in both flow directions, wire raps were introduced as spacers. To replace fuel rods for repair and maintenance, the clusters would be disassembled from foot piece. B4C control rods will be introduced from the core top. Inside of each cluster, among nine moderator boxes, they run five. Fig. 4. Illustrates the cutout view of a single fuel assembly.

The water box and the assembly box are made of stainless steel with sandwich-type construction inside the honeycomb structure mixed with zirconia. It was done to enhance insulation thermally and to get the anticipated rigidity below 0.5 mm deflection to the fuel rods below 500 kPa outside pressure. To reduce the pressure on the honeycomb structure, a venting hole open to the colder side is made. Stainless steel corner pieces are made to minimize the peak stress.

UO_2 fuel with an enrichment ranging from 5% is selected for studies of every thermal assembly design. Waata et al. (2005a, b) considered 5%, and Hofmeister et al. (2005) used a maximum 7% enrichment in their design analysis, which is analogous in the assembly design of Boungiorno (2003). The diameters of the fuel pin suggested in the range from 8 mm outer cladding diameter (Squarer et al. (2003) to 10.2 mm diameter (Yamaji et al. (2005a,b). The suggested design is quite similar to a standard PWR design. To feed adequate mass flux of coolant for the predicted heat transmission, the fuel pins pitch should be quite smaller, despite the considerably smaller coolant mass flow. Cheng et al. (2003) showed in a parametric study that the pitch to diameter ratio fluctuates steadily to achieve the lowest cladding temperature of 1.3 for hexagonal arrangement and 1.15 for square arrangement. To minimize the peak cladding temperatures, Yamaji et al. (2005a,b) applied a pitch to diameter ratio below 1.1 for square fuel pin arrangement.

On the other hand, these small gaps between fuel pins would be dangerous because temperature non-uniformities would cause thermal-

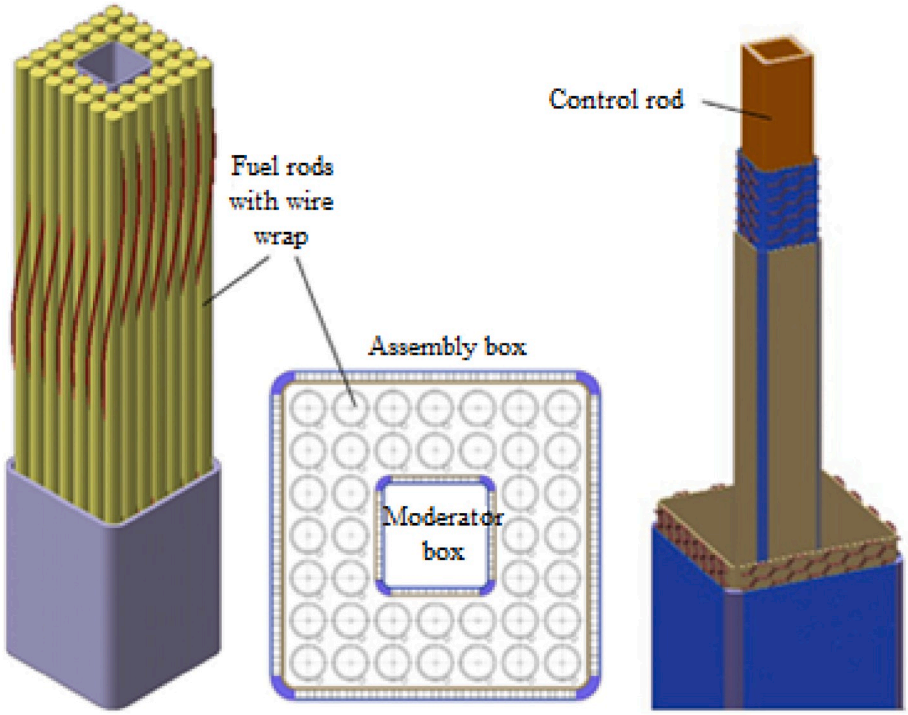

Fig. 4. HPLWR assembly design with wire rapped fuel rods (left) and honeycomb structures of the assembly and moderator box (right). A square control rod is inserted from the top (Courtesy of Schulenberg and Starflinger, 2012).

elastic unsteadiness of the cladding and by the bending of the fuel pin, indicated from a heat transfer studies conducted by Behne et al. (2006). The innovative concept suggested by Bastron et al. (2005) is an artificial roughness of claddings surface, substitute for a smaller pitch of fuel pins. They found a doubled heat transfer coefficient comparable with an axial flow through level claddings, causing the cladding surface temperatures about 50 °C lower at the same mass flow. Besides, inside the assemblies, the coolant merging predicted to be improved. One of the drawbacks of the proposals is that the coolant pressure drop can be raised by about a factor of 8.

The selection of cladding material will depend on the highest surface temperature of cladding allowed for operation. First zircalloy tests showed somewhat higher other alloys with more creep strength along with higher corrosion resistance will be needed to reach 500 °C of coolant temperature. Ehrlich et al. (2004) abridged that stainless steels utilized for sodium-cooled fast reactors such as SS316 or 1.4970 might be the proper solution for creep strength up to 620 °C and reasonable neutron embrittlement. Was and Allen (2005) effectively tested stainless steels and ferritic-martensitic steels regarding corrosion resistance at supercritical water conditions. However, data regarding stress corrosion put concerns at 550 °C by Was et al. (2006) that will expect additional test above 600 °C. Teysseyre et al. (2006) observed a smaller crack depth in Inconel IN690 and IN625, even irradiated IN690 show larger SCC cracks than SS316. Inconel shows less beneficial compared to stainless steels due to higher neutron absorption cross-sections. Oxide dispersed strengthened (ODS) materials might be enabled to design for cladding temperatures beyond 620 °C. However, to verify this assumption, further materials tests will be necessary. In a direct cycle, boric acid does not apply for burn up compensation. More neutron absorption compared to PWR requires to be presumed in control rods. Yamaji et al. (2005a,b) used natural boron carbide (B_4C) in his study, though for compensation of initial reactivity, burnable poisons could be used. Gadolinia (Gd_2O_3) can be applied in several of the fuel rods for this purpose, as suggested by Yamaji et al. (2005a,b) and Joo et al. (2005). A fast or epithermal neutron spectrum could be an alternative to avoid the complication of the required extra moderator at the lower coolant densities. To reach criticality, plutonium accumulation to the fuel of UO_2 with a maximum concentration of 25% would be required.

In contrast, during burn-up, more plutonium would be produced; a sustainable fuel cycle will lead if this plutonium will be recycled and reprocessed. Design studies published earlier by Oka et al. (1995) showed indeed advantageous in comparison with the thermal spectrum. Due to the risk of negative reactivity coefficient of coolant density, this initially fascinating concept has been restricted. Comprehensive core design studies performed by Mori (2005) for such supercritical water-cooled reactors with a fast neutron spectrum for some of the different arrangements of seed and breeding zones, moderator pins (ZrH) in a hexagonal array. However, the danger of a negative reactivity coefficient of coolant density could not yet be omitted.

2.1.2. *Core design concepts*

Most of the core design works have been performed with a thermal neutron spectrum. Core design concepts can be categorized based on flow direction. For single, two, or three-pass core concepts, the flow path varies with the alteration of flow direction at the time of coolant heating. The schematic of the core design with various steps of heat-up is drawn in Fig. 5. The idea of a single-pass core considers a supply of feedwater beneath the core and discharge of hot coolant from the top analogs with PWR. In some assemblies of two-pass core concepts, if downward flow pre-heated the coolant, the mixing plenum at the bottom of the core may be employed to blend non-uniformities of coolant beforehand the next heat up by upward flow, which then performs like a superheater. It will act as a boiler whether the coolant gets heat in three phases. Start with the upward flow of an evaporator, in a steam plenum immediately above the core, a first mixing occurs. Then second heating occurs in a superheater with the downward flow, and finally, an upward flow (a third step) mixed in a second mixing plenum just bottom of the core.

The concept of the single-pass core is similar to PWR with high pressure (25 MPa) and the exit temperature of the core (380 °C). The high-temperature coolant is collected from all fuel assemblies at the core top and then depart from the pressure vessel. Owing to the chosen average exit temperature of 384 °C (at 25 MPa), just lower the pseudo-

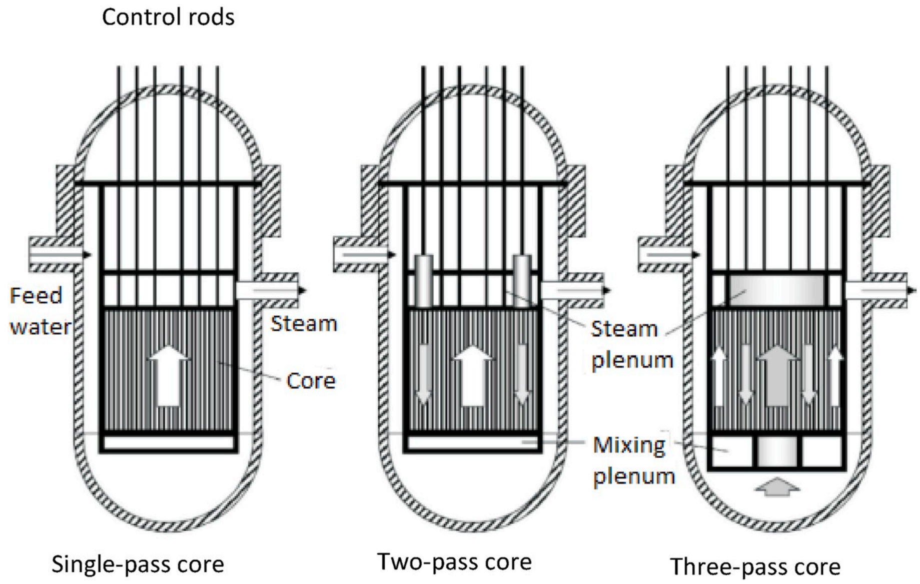

Fig. 5. Core design concepts with multiple heat-up steps (Courtesy of Schulenberg and Starflinger, 2007).

critical temperature, a sub-channel of the coolant at a considerably elevated exit enthalpy is likely to run. This fact contributes to 384 °C water temperature and high specific heat. Vogt et al. (2006), in association with Hofmeister et al. (2005) developed such kind of supercritical core (PWR-SC) using 88 clusters of 2000 MW thermal power. To set a power density of 1000 MW/m^3, coolant with a mass flow of 2772.7 kg/s is heated up between 280 °C to 380 °C like PWR. Here, the highest exit temperature of sub-channel attains only 416 °C. Inside the fuel assemblies and between their gaps, moderator water is passing downwards. The Benefits of this concept are primary loop size reduction, less auxiliary power of the primary pumps, and 2% higher net efficiency compared to current PWR. To eliminate the shortcomings of using an indirect cycle, steam generators, and superheaters in the second loop are required.

To pass the produced steam directly to a high-pressure turbine, the exit temperature of the core should be raised above the pseudo-critical temperature (in case of the direct cycle). To maintain the temperature of the local hot channel within the material limit, the coolant supposed to heat with intermediate mixing in steps. A two-step heat-up (See Fig. 7) is the simplest method recommended by Kamei et al. (2005) and Yamaji et al. (2005a,b). The two-step heat-up concepts named Super LWR with a once-through direct cycle excluding recirculation pumps and steam-water separators. The core comprises of 121 fuel assemblies of a square shape, each with 300 fuel rods and 36 square water rods. Surrounds the fuel assembly, 24 rectangular water rods are added. To achieve about 380 °C in the lower plenum, the coolant flowing downwards is preheated in assemblies of 48 fuels in the core outer section. To distribute the temperature equally, coolant is mixed with downwards moderator water by special moderator rods. The coolant from the lower plenum advances through 73 fuel assemblies in the core inner part that exhibits like a superheater with 500 °C average exit temperature resulting in 2744 MW thermal power. For a more optimized core with a core exit temperature of 530 °C, Yamaji et al. (2005a,b) predicted 732 °C as the highest cladding surface temperature, excluding uncertainties for operation. However, this peak temperature beyond the limits of existing cladding materials. Schulenberg et al. (2007) recommended lowering the exit temperature of the core of about 430 °C to harmonize the creep and corrosion limits of stainless steel.

As per the recommendation by GIF, the exit temperature of the core should be nearly 500 °C to achieve high net efficiency of 44%. Comparable coolant heat-up with an elevation of similar enthalpy can be observed in supercritical fossil power plants. Here, heating of coolant is performed with one evaporator and two superheaters. In an evaporator, the coolant must be heated up with the upward flow and above the core mixed up with a steam plenum. Then the coolant needs to be heated up again in a superheater with the downward flow, and finally, mix with a second mixing plenum just below the core and flow upward direction.

Schulenberg and Starflinger (2012) proposed a probable core design for HPLWR based on the model developed by Hofmeister et al. (2007) to achieve a thermal power of 2188 MW with 156 fuel assemblies and 1160 kg/s mass flow rate. The concept is presented in Fig. 6.

Fuel assemblies and moderator boxes exchange heat to initiate first step-heat-up. The downcomer feed water mixes with moderator water at the lower mixing chamber with an average inlet temperature of 310 °C and then reach to the core center zone. In the inner part of the upper mixing chamber of the evaporator, the coolant temperature rises to 390 °C by 52 clusters. Then downward coolant flows through the superheater 1 of 52 clusters get heat up to 433 °C. The upward coolant after passing through superheater 2 raises the temperature to 500 °C by another 52 fuel assemblies in the core-periphery. This design remains within the available cladding materials limit (proposed peak cladding

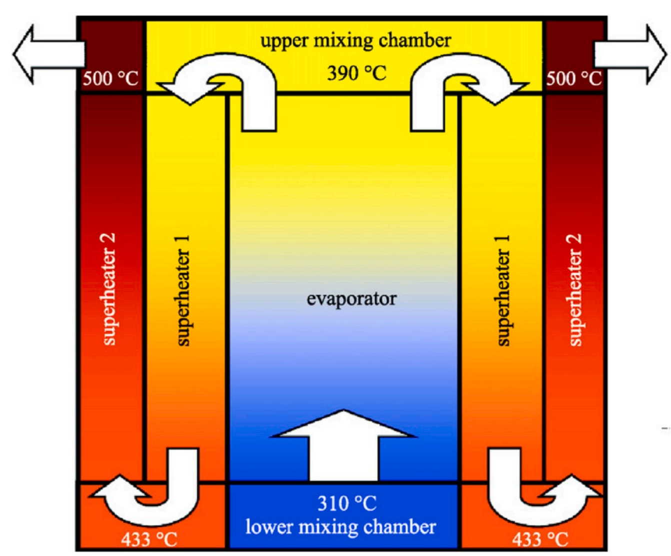

Fig. 6. Three pass core concepts with predicted temperatures for each heat-up step (Courtesy of Schulenberg and Starflinger, 2012).

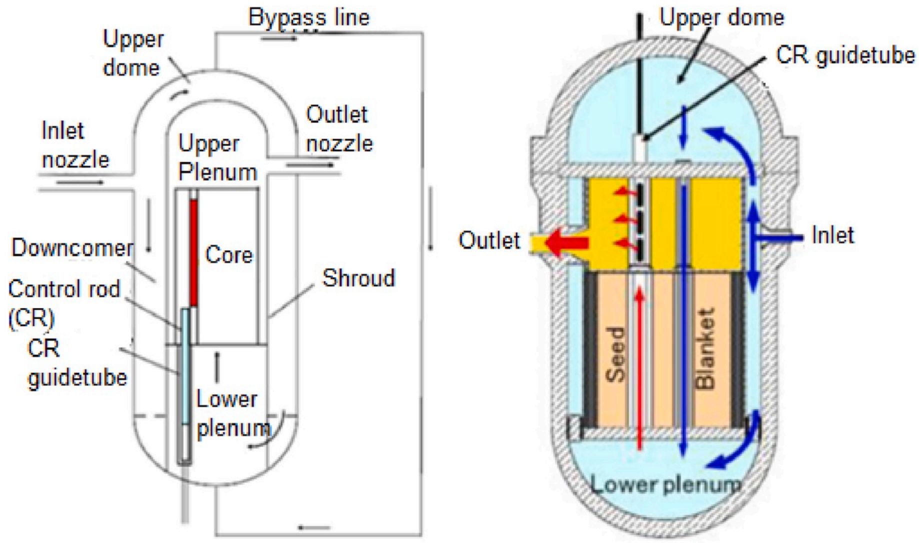

Fig. 7. Configuration of Japan SCWRs: thermal spectrum (Courtesy of Yamada et al., 2011; fast spectrum SCWR; Courtesy of Nakatsuka et al., 2010).

temperature is 620 °C).

2.2. Japanese SCWR design concepts

Waseda University, in collaboration with Tokyo University, is developing the SCWRs using the pressure-vessel design concepts called Super LWR and Super FP using the thermal neutron spectrum and fast neutron spectrum, respectively. Suitable R&D technologies to both ideas of SCWR include thermal hydraulics, water chemistry, and materials research. The design objective of the development of Super LWR concepts is to meet-up the challenges of the market economy with the reduction of capital cost. On the other hand, the design objective of the development of the Super FR concept is to lower the price than present LWR or the Super LWR. The configurations of the thermal and fast neutron spectrum core with flow path (in Japan) are illustrated in Fig. 7.

2.2.1. Core design concepts with thermal neutron spectrum

The core design of Japanese SCWR has been studied conceptually. First, Oka et al. (1992) designed a core with a fixed moderator of zirconium-hydride rods following the thermal neutron spectrum with a cladding material of stainless steel. The temperature of the core outlet reached about 400 °C. Safety criterion of stainless steel cladding was satisfied by Koshizuka et al. (1994) by analyzing the loss of coolant accident (LOCA). Because of the simplicity and the reduction of radioactive waste after use, water rods were considered better than zirconium-hydride. Next, Okano et al. (1994) compared the core characteristics from the neutron moderation with the help of water rods of single, semi-double, and full-double type. The core with the full-double tube water rod was preferable for minimal reduction of coolant density at the core top. Okano et al. (1996) designed the SCWR core using water rods of a full double tube. SCLWR core was developed by Dobashi et al. (1997) using water rods of the single tube (SCLWR-SWR) along with its steam cycle. This core reaches the core outlet temperature, thermal efficiency, and electric power of 397 °C, of 40.7% and 1013 MW, respectively. To reduce the thermal fatigue issue of the control rod guide tubes, Tanaka et al. (1996 studied the SCLWR-SWR core and suggested a core with the downward flow in the water. Studies of Oka et al. (1995, 1992), Oka and Koshizuka (1993, 1996), Koshizuka et al. (1994), Okano et al. (1994, 1996), Tanaka et al. (1996), Dobashi et al. (1997) showed the outlet temperature of SCLWR cores of around 400 °C using cladding materials of stainless steel. The maximum surface temperature criterion was 450 °C.

In Japan, three decades ago, thermal power plants with supercritical fluid reached the outlet temperature of 566 °C. Dobashi et al. (1998) designed a higher outlet temperature core (SCLWR-H) using the cladding material of Ni-based alloy. A Water rod of a single tube is adopted to make the structure simple (Okano et al., 1994). The temperature core outlet and the efficiency (thermal) attained high at 508 °C and 44%, respectively. The electric power reached at 1610 MW.

The University of Tokyo proposed a Super LWR concept with a two-pass core, which is approved worldwide (Yamaji et al., 2005a,b; Oka and Koshizuka, 1996; Kamei et al., 2006). It is illustrated in Fig. 8(a). In the reactor pressure vessel upper dome, a portion of the inlet coolant is fed that channeled to the water rods through distribution tubes. The other part of the inlet coolant passes across the downcomer and merges with the first portion at the bottom dome. Then the mixing water flows up through the fuel channels to the upper mixing plenum. The outlet temperature (average) reaches 500 °C. Because of low-temperature moderator mix up in lower plenum. The high-density moderator can efficiently balance the upper core moderation in the water rods. However, the upper structure of the two-pass core is complicated because of the moderator guide tube/distribution tube, with the fuel assemblies and the ties between interfaces of heated and cold coolant. The refueling system is also complicated, which is required to test lower plenum coolant mix up. Wu and Oka (2014) carried out a core design study to make the upper core structure simple. The flow pattern is illustrated in Fig. 8(b). The flows of light water downward through the inlet nozzle of the lower dome and pass the inner water rod and lower plenum. The inner water rod moderator flows with changes of direction at the top core outer water rod. After that, the moderator passes out from the bottom mixing plenum to the base core and then combined with the light water of the bottom dome. The finely mixed coolant flows finally to the upper plenum through the fuel channel. The upper core structure is thereby made simple by eliminating guide distribution tubes thought the bottom plenum is still complicated. Wu et al. (2013) studied at Waseda University the single-pass coolant flow pattern at low coolant temperature core, aiming to simplify the structure of the upper core. The flow distribution is shown in Fig. 8(c).

The flows of inlet water descending to the lower dome and emerge through fuel channels and the water rods. The moderator guide tubes at the upper core are omitted, and the bottom plenum is similar to PWR. Considering the same design criteria, the low-temperature core is designed except maximum cladding surface temperature limitation as that in the core of the two-pass system. From the evaluated relationship, it is realized that with the increase of the maximum cladding surface temperature, the outlet temperature increases linearly. When the

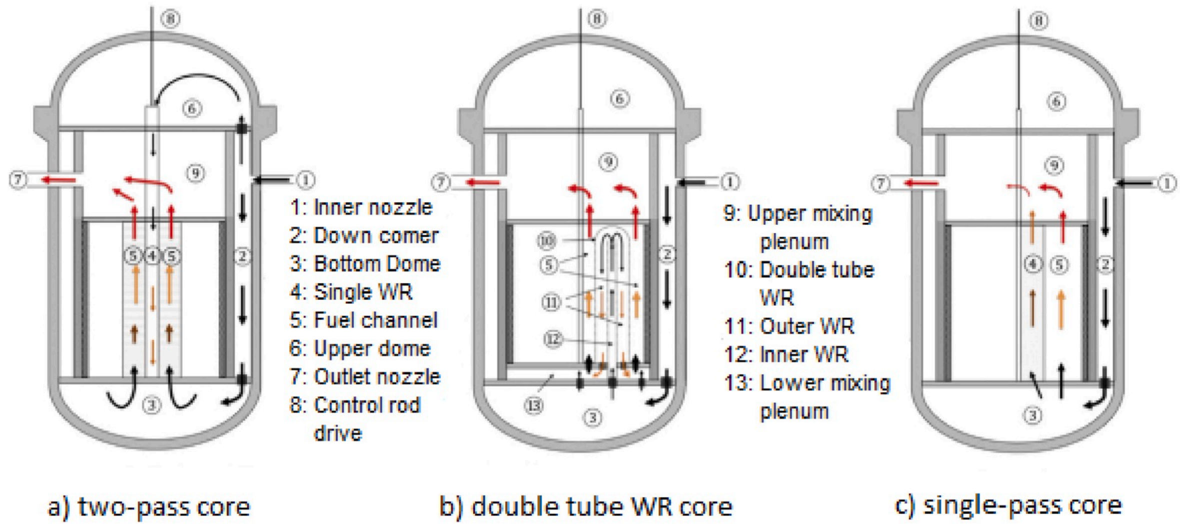

Fig. 8. Flow scheme for different cores (Courtesy of Wu and Oka., 2014).

maximum cladding surface-temperature is 650 °C, the outlet temperature is 465 °C. Nevertheless, to use a high-pressure steam turbine of traditional supercritical FPP, of which the highest temperature of the steam is 538 °C, outlet temperature increase to 500 °C at the present maximum cladding surface temperature criterion is crucial. The core design criteria are presented in Table 2. The core design criteria are selected based on similar design goals, excluding outlet temperature (average) as earlier low-temperature single pass core proposed by Wu et al. (2013). To guarantee the integrity of fuel rod and safety of core, the same design criteria at regular operation is used.

In Table 3, the main features of low and high single-pass core design are abridged. The local power peaking distribution over the core is improved by adopting separation plates and radial enrichment zoning in peripheral and inner assemblies, respectively. Consequently, the average outlet temperature is increased to 500 °C with cladding maximum surface temperature of 656 °C. The efficiency (thermal) increases from 43.1% to 43.8% with low-temperature single pass core. In contrast, the discharge burn-up is also enhanced and achieved to 45.3 GWd/t by applying five fuel batches.

2.2.2. Fuel assembly design concepts with thermal neutron spectrum

In the past years, extensive R&D activities have been carried out considering different features of the development of SCWR. Among the R&D activities, the design of the fuel assembly is a crucial task. In this regard, many fuel assembly designs of SCWR have been suggested. Dobashi et al. (1998) and Cheng et al. (2003) proposed hexagonal arrangements among the fuel assemblies of thermal-spectrum SCWR design. Oka et al. (2002) and Cheng et al. (2003) have been proposed a square arrangement of fuel assemblies. Boungiorno and MacDonald (2003) and Bae et al. (2004) have proposed concepts of a solid moderator. The neutron kinetics and thermal-hydraulic analysis have been performed separately in the works mentioned above. However, because of the strong difference in the thermal-hydraulic characteristics, especially in the density of supercritical water, a special method that couples the neutron physical analysis with the thermal-hydraulic analysis was adopted.

Yamaji et al. (2003) developed a basic concept of an SCWR core design using a coupled neutronic model (three-dimensional) and thermal-hydraulic model (single channel). Nevertheless, the detailed distributions of thermal-hydraulic parameters in various sub-channels cannot be assessed through this method. Waata et al. (2005a,b) have developed a coupled system for analysis of fuel assembly with the help of Monte Carlo code MCNP and sub-channels code STAFS. Though this system provides local characters of fuel rods and sub-channels, large computation efforts are extremely required. Recently, Liu and Cheng (2009) have been developed a fuel assembly design for a thermal SCWR.

Table 2
Core design criteria (Kamei et al., 2006).

Criteria for thermal design
Maximum linear heat generation rate at rated power ≦ 39 kW m^{-1}
Maximum cladding surface temperature at rated power ≦ 650 °C for stainless steel cladding
Moderator temperature in water rods ≦ 384 °C (the pseudo critical temperature at 25 MPa)
Neutronic design criteria
Positive water density reactivity coefficient (negative void reactivity coefficient)
Core shutdown margin ≧ 1.0%ΔK/K
Criteria for other design
Simple upper core structure
500 °C average outlet temperature for ensuring 43.8% thermal efficiency.
Electric power scale over 1000 MWe.
Average discharge burn-up ≅45 GWd/t.

Table 3
Main features of high and low-temperature cores (Wu and Oka, 2014).

Cores	Low-Temperature core	High-Temperature core
Thermal power (MW)	2804	3492
electric power (MW)	1200	1530
Thermal efficiency (%)	43.1	43.8
Operating pressure (MPa)	25	25
Temperature inlet/outlet (°C)	280/465	280/500
MCST (°C)	650	656
Number of fuel assembly	121	129
Ave. fuel enrichment (%)	7.30	7.31
Fuel batch numbers	4	5
Gd_2O_3 concentration (inner/ peripheral assembly) (%)	4%/4%	8%/1%
Fuel rod dia./pitch (cm)	UO2/0.8/0.9	UO2/0.8/0.9
Cladding/thickness (cm)	SS/0.05	SS/0.05
Average power density (MW/m^3)	93.4	97.6
Core effective height/diameter (m)	3.70.3.23	4.20/3.31
Discharge burn-up (GWd/t)	43.3	45.3
Maximum linear heat generation rate (kW/m)	38.5	37.4
K-eff (BOC)	1.1230	1.1040
K-eff (EOC)	1.0023	1.0005
Shutdown margin (%dk/k)	2.99	1.45
Application of separation plates	No assemblies	In peripheral assemblies

The fuel assemblies have two rows between the moderator channels compared to the existing fuel assemblies to achieve a uniform moderation for every fuel rod cells. Consequently, power distribution (radial) becomes uniform.

The geometric arrangement of the fuel assembly of a thermal SCWR is sketched in Fig. 9. In this fuel assembly designs have only one-row fuel rods between two moderator channels) Cheng et al., 2003; Oka et al., 2002). Channels of two moderators bound every inner fuel rod. On the other hand, adjacent to the assembly wall fuel rods are surrounded by neighboring single moderator channels only. This provides the interior region a stronger moderation, and consequently an elevated power density. As a result, it possesses a strong non-uniform radial power density distribution. The power density difference between fuel rods might be reached to 40% (Liu and Cheng, 2009). Liu and Cheng (2009) therefore, proposed a new design of fuel assembly with improved radial power distribution. Fig. 10 represents the geometric arrangement of the proposed fuel assembly. Between the channels of the moderator, fuel pins are arranged in two rows, every single fuel rod is faced towards a single moderator channel. In this way, the entire cross-section could be achieved a more identical moderation. Several major parameters of the proposed fuel assembly are summarized in Table 4.

A larger moderator channel is chosen considering the fuel rods of two rows, each of which arranged in 4 × 4 fuel rods. This moderator channel will provide a sufficient moderation capacity. The fuel assembly with high mass flux is needed to enhance the heat transfer between cladding and coolant, where a ratio of small pitch/diameter is required. In contrast, Cheng et al. (2007) showed that a significant non-similar local heat transfer is led due to a small value of pitch/diameter ratio. Thus, the selected pitch/diameter ratio of 1.2 seems to be a reasonable value. Moreover, nine grid spacers are arranged uniformly along with the fuel assembly. A gap of 1.0 mm between fuel assemblies is chosen following the current PWR design considering the thermal expansions and neutron-induced swelling with a radial temperature or gradient of neutron flux or the refueling process.

2.2.3. Core and fuel assembly design concepts with fast neutron spectrum

The super FR developed by the University of Tokyo is a fast spectrum SCWR, which requires less moderation (Yoo et al., 2006). As a result, for economic competitiveness, the core is designed to design in a compact size with a reasonably high power density (Liu and Oka, 2013a) Decreasing fast reactors capital cost less than LWRs is a prime goal for closed fuel cycle (Schulenburg et al., 2014). The volume of spent fuels, radioactive waste, and recycle plutonium are reduced in fast reactors. However, capital cost reduction is still a challenge to commercialize (Schulenburg et al., 2014).

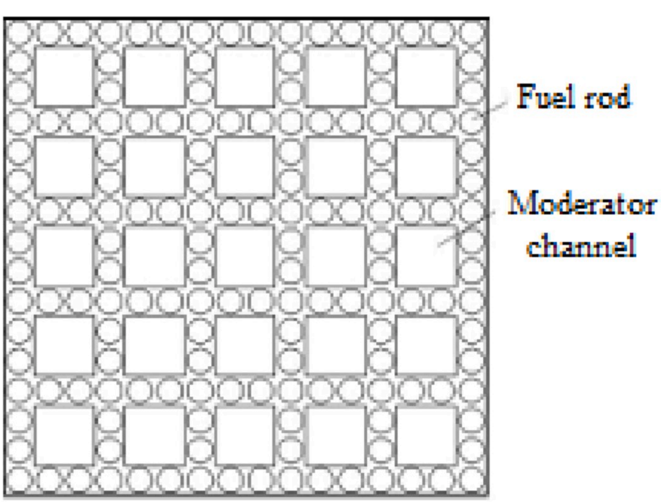

Fig. 9. Sketch of a thermal SCWR fuel assembly (Courtesy of Liu and Cheng, 2009).

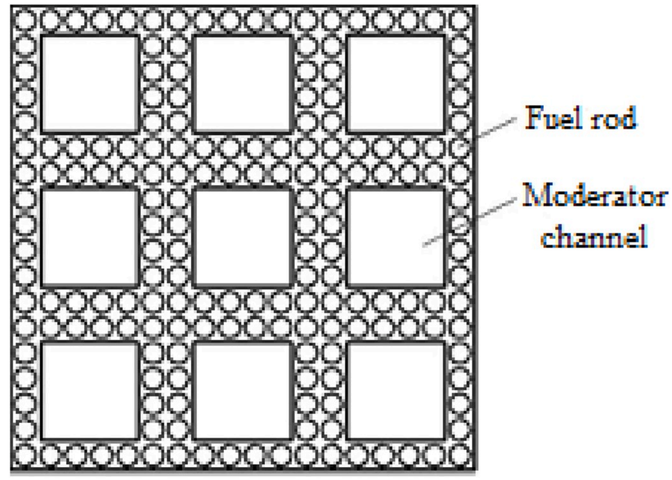

Fig. 10. Sketch of the new SCWR fuel assembly (Courtesy of Liu and Cheng, 2009).

It was shown by previous design studies (Juvremovic et al., 1993; Oka and Juvremovic, 1996) that Super FR provides zirconium-hydride thin layers in between seeds and blankets caused a negative void reactivity of coolant. Recent studies proposed MOX fuel assembly of tight lattice in hexagonal geometry with few blankets inserted in the core (Yoo et al., 2006; Cao et al., 2008). Depleted UO_2 fuel and ZrH solid moderator is used in the blanket assembly. During the loss of coolant, the possible positive void reactivity coefficient is one crucial feature and design challenge. Therefore, the blanket assembly compensates for the reactivity loss with burn up and acts as a neutron absorber and moderator.

Various flow patterns with downward flow blanket assembly have been suggested as the first path, and all seeds assemblies with up-flow are in the second path. Yoo et al. (2006) showed that in the blanket assembly, the presence of high-density coolant would assist in creating more axial uniform coolant density distribution. Cao et al. (2008) showed in their improved design that several seed assemblies are relocated from the second path to the first path to develop the characteristics of the core during accidents and transients. Ikejiri et al. (2010) proved by safety analysis that during the accident of the complete loss of flow, this design is useful to reduce the highest surface temperature of the cladding in the second path. It also assists to continue the flow from the seed assembly of the first path with coolant expansion. Nevertheless, from the flow instability aspect and manufacturing technology, some issues are remained, specifically in the process of LOCA and re-flooding. The direction of flow and the buoyancy are the opposite. During the

Table 4
Reference parameters of the two-row FA (Liu and Cheng, 2009).

Reference Parameters	Value
The design parameters	18 × 18
Fuel rods no. in each fuel assembly	180
Side of the fuel assembly (mm)	177.2
Fuel assembly box wall thickness (mm)	2.0
Ratio of Pitch-to-diameter (–)	1.2
Fuel rod Diameter (mm)	8.0
Fuel rod cladding thickness (mm)	0.5
Moderator channel wall clearance (mm)	1.0
Moderator channel wall thickness of (mm)	0.5
Mass flow fraction of moderator (%)	30
Moderator to fuel volume ratio (–)	3.06
Power density (MW/m^3)	110.2
Linear power (Average) (kW/m)	19.2
Fuel rod Enrichment (%)	4.0
Cladding, water rod, and assembly wall material	Alloy 718

accidents, flow and heat transfer will be obstructed. Moreover, the upper core structure of the in-vessel upper core structure and control rod element; the flow path is very complicated in the downward. Ishiwatari and Wu (2011) by the safety analysis has been suggested an upward flow scheme which can meaningfully make the upper structure of the core simple and thereby more practicable for mechanical improvement of the control rod. In this connection, Liu and Oka (2013b) proposed the design of 1000 MWe upward flow super FR core. The proposed development includes two upward flow schemes. MOX fuel pins of the compact lattice are organized in seed assembly hexagonally and fuel pins of depleted UO2 with the ZrH layer wrapped in blanket assembly to introduce negative void reactivity. This core design comprises seed and blanket assemblies of 162 and 73, respectively. The core active height is 3.6 m and ZrH layered solid moderation of 1.7 cm. The axial core coolant density distribution away from uniform caused by the upward flow. Uniform power distribution is achieved using axial Pu enrichment and fresh loading and refueling patterns. The design criteria with up to 237 kW/L, 500 °C average core outlet temperature have been fulfilled with all limitations, whole core, and local negative void reactivity. The in-vessel coolant flow pattern, specification of fuel rod and seed assembly, and the blanket assembly and are shown in Figs. 11–13, respectively.

Previously, the flow pattern of two path coolant is considered in the Super-Fast Reactor. In the latest core design, one pass is considered. Fig. 14 shows the Super-Fast Reactor's substitute flow scheme for one path core. In the single pass-core, in-core structure and refueling are simplified by removing the coolant seals in between hot and cold (Wu and Oka, 2014). By implementing a MOX fuel area beneath the assembly of fuel blanket, the power change with burn up is mitigated. Fig. 15 illustrates the fuel assemblies 'cross-section. Fig. 16 depicts the fuel assemblies loading scheme in the core. For better negative void reactivity, seed and blanket assembly's array is proved heterogeneously. To get power distribution radial and flat, the fuel shuffling pattern in three bath types is performed in case of seed assemblies, while in every cycle, assemblies of the new blanket are refueled. The rate of flows adjusted separately by inlet orifices for every assembly and keeps fixed through the burn-up period. To fulfill the requirement for the maximum temperature of the cladding surface during the burn-up period, the loading pattern is optimized. Due to the channel box in between assemblies, no flow mixing is provided. Table 5 shows the summarization of the features of the single-pass core Super-Fast Reactor.

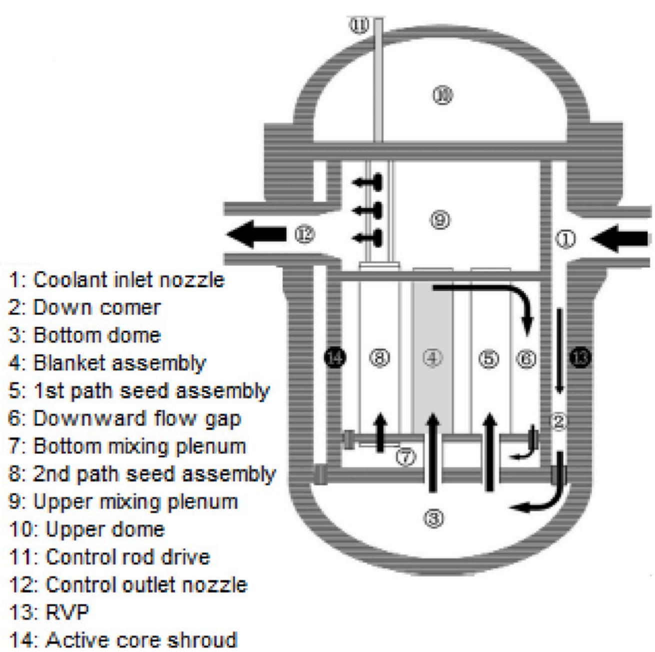

Fig. 11. In-vessel coolant flow pattern (Courtesy of Liu and Oka, 2013b).

2.3. Canadian SCWR design concepts

The Canadian SCWR concept, which is a generation IV reactor, is developing by Atomic Energy Canada Limited (AECL). The reactor has been developed from the well-proved CANDUÒ reactor of pressure tube type (Yetisir et al., 2016). The primary use of Canadian SCWR is the production of electricity. Potential supplementary products of the SCWR include the hydrogen, treated heat, and desalination water.

Canadian SCWR is a pressure tube type SCWR that has many advantages as compared to the pressure vessel type SCWR. It comprises of 10–15 mm thickness distributed pressure channels, while the pressure vessel type reactor has a large pressure vessel of wall thickness 0.5 m. The pressure tube type reactor design is considered more adjustable concerning flux, flow, and density changes compared to pressure vessel type reactors. Canadian SCWR incorporates a heavy water moderator with a thorium fuel cycle to boost-up safety features. Fig. 17 depicts the Canadian SCWR's cross-sectional view with distributed pressure tubes (Yetisir et al., 2018).

The Canadian SCWR is developed to generate a thermal power of 2540 MW and an electric power of 1200 MW considering 48% thermal efficiency of the plant. Water enters at ~350 °C and exits at 625 °C with a pressure of 25 MPa. The flow scheme of the Canadian SCWR core is also shown in Fig. 17. The coolant passes through the inlet nozzles to the inlet plenum and then inters into the fuel conduit. In spite of the high pressure and relatively low coolant temperature, a plenum is suitable for the core inlet. During reloading of the fuel, the inlet plenum top can be removed. The tube sheet of bottom inlet plenum is a square array of holes; the size of the hole is the same as the pressure tube. At the proposed higher and temperature, the present on-line refueling practice of CANDU practice is too crucial, and a batch refueling strategy is implemented. 336 fuel channels arranged in a 5 m long fuel assembly accommodated in the Canadian SCWR core. The thermal power of average fuel channels 7.56 MW, and the core radial power peaking factor is approximately 1.32.250 mm lattice pitch of the channels is considered to achieve a negative void coefficient and high burnup (MacDonald et al., 2015). The CANDU standard pressure tube design cannot be applied to Canadian SCWR because of the coolant high pressure and temperature.

Several design ideas for the Canadian SCWR fuel channel have been presented by Bushby et al. (2000) and Dimmick et al. (1998). Two design concepts: the high-efficiency fuel channel and the re-entrant fuel channel proposed by Chow and Khartabil (2008). Afterward, the fuel channel is modified and latest design of fuel channel and fuel assembly is presented by Yetisir et al. (2018) describes in this section.

2.3.1. Fuel channel and fuel assembly

The fuel concepts of Canadian SCWR are different from other SCWR. The fuel of the Canadian SCWR is accommodated within the fuel channel, detaching the coolant from the moderator. The Canadian SCWR core consists of 336 fuel channels of power generation capacity 2540 MW (Thermal). The fuel channels number is considered to be multiple of 12, to maintain the quarter symmetry arrangement of fuel reshuffling at three positions.

Fig. 18 shows a schematic diagram of a fuel channel and a fuel assembly of Canadian SCWR. Every fuel channel comprises a pressure tube that is immersed in heavy water and includes a replaceable fuel assembly. The fuel assembly contains a fuel bundle (64-element two-ring type), a central flow tube, an encapsulated (by inner and outer liners) ceramic insulator, along a locking mechanism to permits the fuel assembly fast installation and removal. Fig. 18 (a) comprises a pressure tube coupled with the tube sheet in a leak-tight manner, a pressure and tube extension guide tube that expands the fuel channel to the inlet plenum, and an enlargement bellows that joins the guide tube to the outlet header. The penetration holes in the tube sheet and at the outer

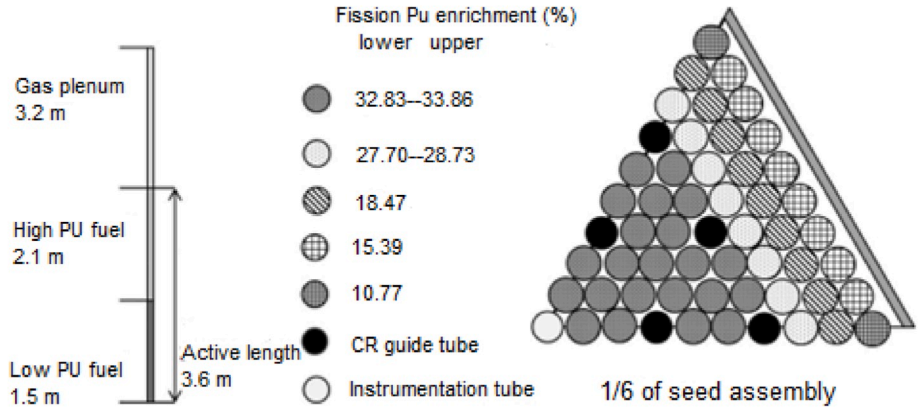

Fig. 12. The specification of fuel rod and seed assembly (Courtesy of Liu and Oka, 2013b).

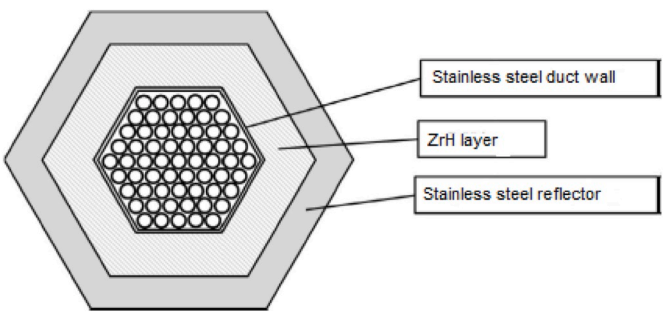

Fig. 13. The specification of blanket assembly (Courtesy of Liu and Oka, 2013b).

header considered that all components of the fuel channel are possible to detach. This orientation allows the replacement of single fuel but also a complete core. Fig. 18 (b) represents the fuel assembly, contains all in-core components experienced in high-level radiation areas. For a maximum of 3 operating cycles, the fuel assembly remains in the core. Fig. 18 (c) depicts that fuel assembly is installed in the fuel channel. To adapt manufacturing tolerances and to simplify the fuel assembly installation and removal, an annual gap is maintained between the fuel assembly and the pressure tube. In the fuel channel concept of Canadian SCWR, this gap is a critical feature to meet the demand of no-core-melt. This gap is also optimized to reduce heat conductance in normal operating situations. In case of accidental conditions, rising the temperature result in the expansion of fuel assembly and consequently reduce the gap size. The decay heat transfer from the core to the moderator will be more effective.

2.3.2. *Small modular pressure tube SCWR*

Recently, Yetisir et al. (2012) proposed a small modular version pressure tube SCWR called Supersafe© based on the preliminary concept of the Supersafe© reactor proposed by (Duffey et al., 2011) which incorporates the enhanced safety, improved economics, increased security, and enhanced sustainability compared to current reactors. It is a smaller version of Canadian SCWR with an output power of 300 MWe, pressure 25 MPa, and 625 °C fluid outlet temperature with a cycle efficiency of 45%. The concepts of Supersafe© reactor include the improved features which could be implemented in a less populated area where a full Canadian SCWR is unrealistic to achieve. The supercritical turbine technology is similar to those in present fossil-fired power plants using supercritical water. In calandria vessel of low-pressure, fuel channels are submerged within a pool of subcooled heavy-water moderator. Each fuel channel consists of a pressure tube and a ceramic insulator, which is enclosed within a sleeve of porous stainless steel. The moderator ensures cooling of the core for all scenarios like

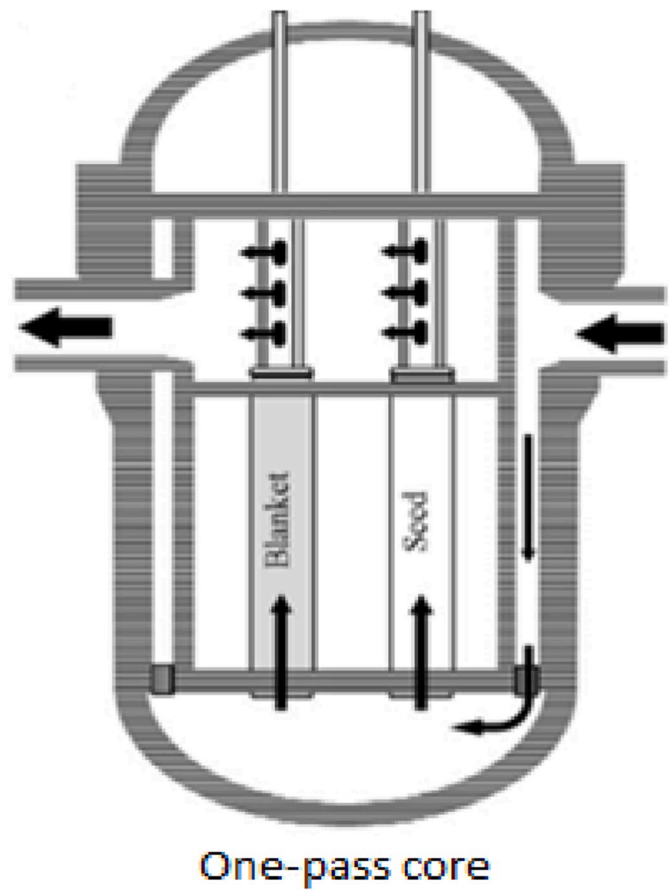

Fig. 14. Substitute flow scheme of one path core Super-Fast Reactor (Courtesy of Schulenburg et al., 2014).

normal operation and postulated accidents. This design characteristic allows a flash-driven passive moderator cooling as an inherent safety feature. The main objective is to attain "no core melt" in passive mode.

2.3.3. *The reactor core design*

Fig. 19 shows the cross-sectional views of Supersafe© reactor concept schematically. In a pressurized inlet plenum at the core top, a light water coolant is enclosed. The core is attached to the bottom of a calandria vessel with a heavy water moderator (Fig. 19-A), fuel assembly placed inside the pressure tube surrounded by the moderator, and vertically oriented pressure tubes connect with a tube sheet (Fig. 19-B). The tube sheet keeps isolating the coolant from the moderator. After the

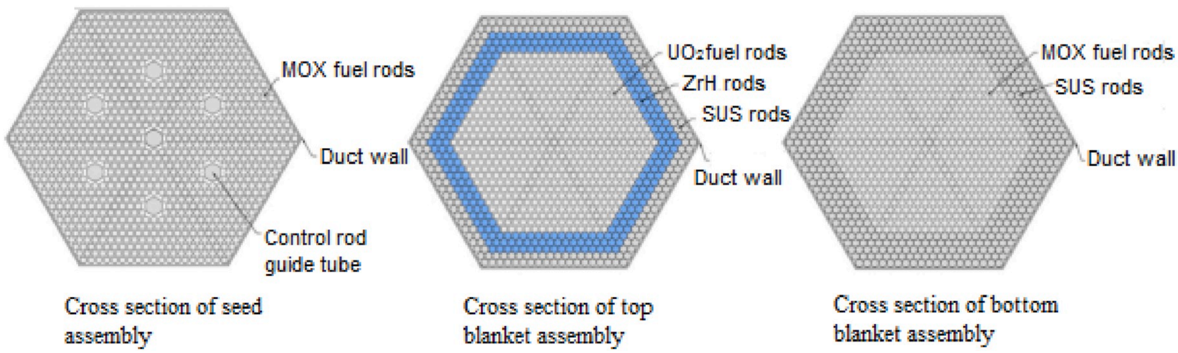

Fig. 15. Cross-section of an assembly of the one path core of the Super FR (Courtesy of Liu and Oka, 2013b).

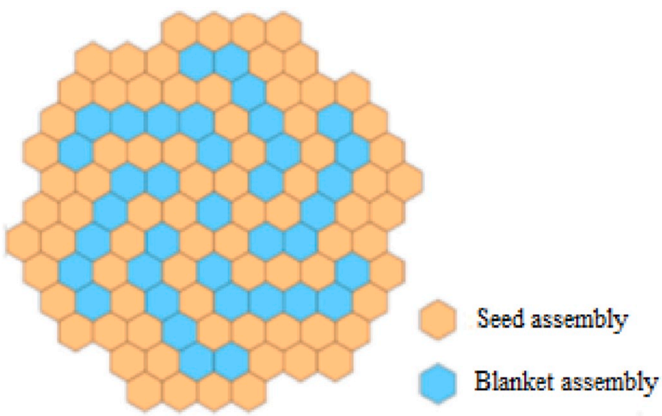

Fig. 16. Loading pattern of the one path core of the super FR (Courtesy of Liu and Oka, 2013b).

Table 5
Super-Fast Reactor features (Liu and Oka, 2013b).

Power MWt/MWe	2337/1006
Coolant pressure (MPa)	25.0
Inlet/outlet temperature (^{o}C)	289/501
MCST in seed (^{o}C) BOEC/EOEC	646/647
MCST in blanket (^{o}C) BOEC/EOEC	560/647
Active power density (kW/L)	206
Overall power density (kW/L)	149
No. of seed assemblies	78
No. of blanket assemblies	37
Height of the active core (m)	2.4
Eq. diameter of the active core (m)	2.47
Plutonium enrichment in seed assembly (wt%)	32 (bottom)/25 (top)
Plutonium enrichment in bottom blanket (wt%)	10 (bottom)
Cycle length (EFPD)	200
Fuel batch	3
Ave./max discharge burn-up (GWd/t)	53.8/72.7

three-batch cycle, the reactor is refueled (Yetisir et al., 2012). Fig. 19 also depicts the coolant and moderator flow streams in a simplified view. The coolant flows from inlet nozzles to inlet plenum (Fig. 19-A) and finally emerges into the fuel channel (Fig. 19-C). The coolant exits the central flow tube and is redirected upwards through the fuel elements (Fig. 19-D). The coolant of the inlet plenum is initially at a subcritical temperature of 350 °C, at a pressure of approximately 26 MPa (Leung et al., 2011). The coolant continues subcritical unless heating up on its upward pathway inside the fuel channel. The supercritical water streams inside the inlet plenum ((Fig. 19-A and C), mix with that in the outlet plenum through the fuel channels. Cladding temperature keeps fixing at a tolerable level to get high efficiency. Exit temperature is maintained at about 625 °C with the help of channel-specific orifices.

120 fuel channels accommodated in the core of the Supersafe© reactor and placed in a 25 cm lattice pitch of square size. The core diameter is 400 cm, including a region o radial reflector of 50 cm thick D_2O. In this concept, the axial reflector region is not included. Nevertheless, to give an extra reflection of neutron and to get an axial flatter power profile, the moderator of the heavy water might be enlarged 50 cm below the lowest and above the highest fuel elevations. The Supersafe© reactor uses the same fuel assembly and channel designs as that of the Conceptual fuel channel design for Canadian SCWR. The specifications of the assembly and the channel of fuel are derived from Pancer et al. (2012). The fuel channel of the high-efficiency concept is considered in the Supersafe© reactor, which has been described by Chow and Khartabil (2008). The studies on the optimization of the fuel assembly and the fuel channel are in progress (Yetisir et al., 2012). Due to the coupling of thermal-hydraulics, physics, and the conceptual design of mechanical aspects, an iterative approach is required. The intended parameters like the fuel assembly of the average core, fuel burnup, fuel enrichment, coefficients of reactivity, power peaking factors of both axial and radial, temperatures of fuel and cladding, reactivity, stability, and linear power rating are considered optimization (Yetisir et al., 2012).

2.3.4. Options of the fuel cycle

Once through cycle, Pu-Th based fuel is the reference fuel for the conceptual fuel channel design for Canadian SCWR (Magill et al., 2011). The cycle is chosen as there is no requirement of enrichment with U-235 and, therefore, preserving resources of natural uranium and thus assisting in fulfilling the GIF's goals of sustainability. The once-through cycle will be used in the SCWR, as long as an ample reserve of U-233 in spent fuel is piled up that can be employed in a recycling-based U-233 fuel cycle (Magill et al., 2011). For example, the Pu-Th based fuel cycle relies on availability and Pu recycling from the spent fuel of PWR. Yetisir et al. (2012) proposed that in the short term, employing a different fuel cycle in the Supersafe© reactor may be desirable biasing on the program to economic, deployment, or functional limitations. For comparison to the reference fuel cycle, two simple fuel cycles, such as low enriched uranium (LEU) and an LEU-Th-based cycle, are investigated. For the three fuel cycle options, a comparison of parameters of the fuel cycle is presented in Table 6. Both of the cycles are once-through the cycle, driven by initial enrichment of U-235, and fuel compositions in both within the fuel assemblies and within the Supersafe© reactor are distributed uniformly. In order to compare options of alternative fuel with the reference Pu-Th fuel cycle, the LEU with fissile enrichments and LEU-Th were attuned with the intention that all three fuel cycles arrive at the equal target exit burn-up in the supersafe© reactor.

2.3.5. Refueling of SCWR

Refueling issues has an imperative influence on the SCWR concept. Refueling activities are usually on the critical path for returning the nuclear reactor to operation. Therefore, the procedure of refueling and

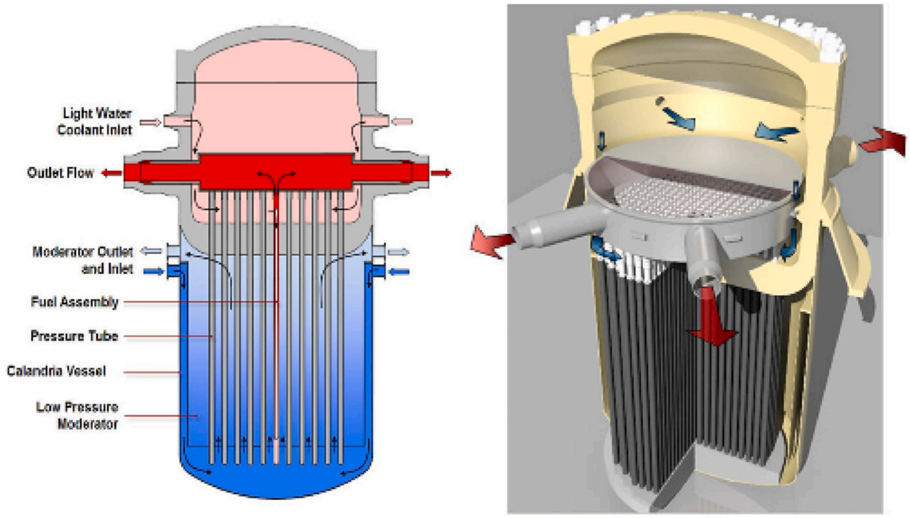

Fig. 17. The core design of Canadian SCWR (Courtesy of Yetisir et al., 2018).

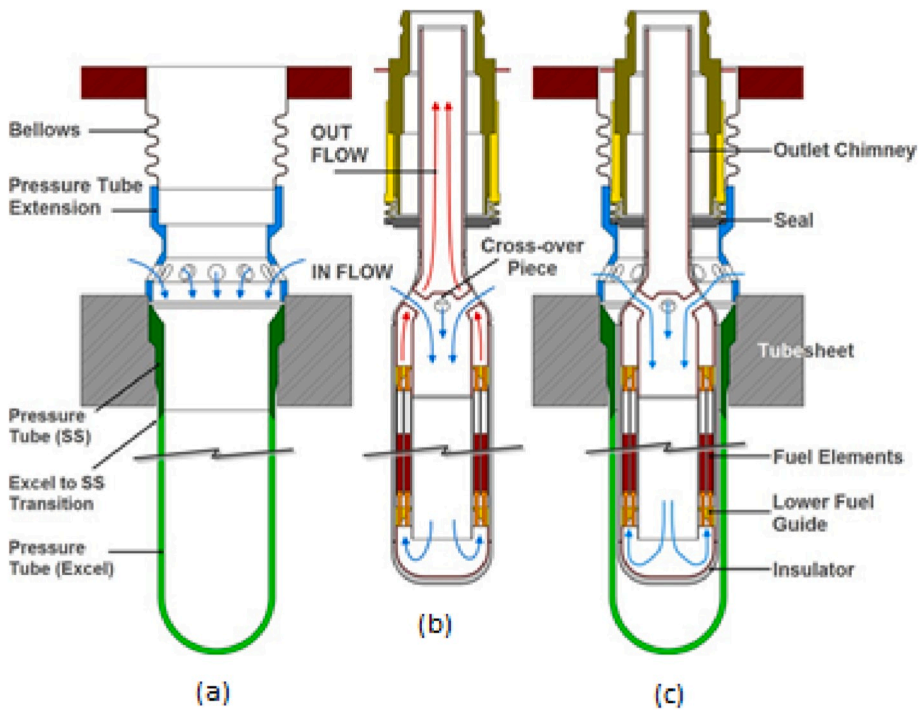

Fig. 18. Fuel channel concept of Canadian SCWR: (*a*) schematic of fuel channel, (*b*) Schematic of fuel assembly, and (*c*) Schematic of loaded fuel channel with fuel assembly (Courtesy of Yetisir et al., 2018).

fuel handling tasks is a vital economic issue for SCWR. Additionally, reducing radiation exposure to the workers and decreasing the release of fission products because of fuel failure are two essential objectives that must be taken into account in fuel designing, fuel fabrication, fuel bundle manufacturing, and spent fuel storing. Batch refueling procedure has been used by Canadian SCWR. After a fuel cycle (425 days for Canadian SCWR), the core is rearranged, with one-third of the core substituted. The spent fuel pool accommodates 1584 fuel assemblies. To limit the probable spreading of contamination, faulty fuel is stored and handled discreetly. In a capacity of 120-fuel assembly pit, new fuel is dry-stored. During refueling, a buffer pool in the reactor building permits temporary storage of 112 new and 56 spent fuel assemblies. The buffer pool is connected to the spent fuel pool by an inclined fuel transfer machine and a horizontal transfer system. Through the inclined transfer channel and horizontal channel, a four fuel assemblies' capacity fuel magazine is used for transferring the fuel assemblies. The refueling machine comprises a camera, a distantly controlled robotic arm accompanied by a telescoping pole and pendant. A fuel magazine is supposed to reduce the needed commute numbers between the reactor vessel and fuel racks. In fuel racks construction, neutron-absorbing materials are utilized to ensure subcritical of stored new fuel in case of flooding. Japanese Supercritical Water-Cooled Reactor applies the system the same as the current BWR's refueling machine with some changes for longer and heavier fuels.

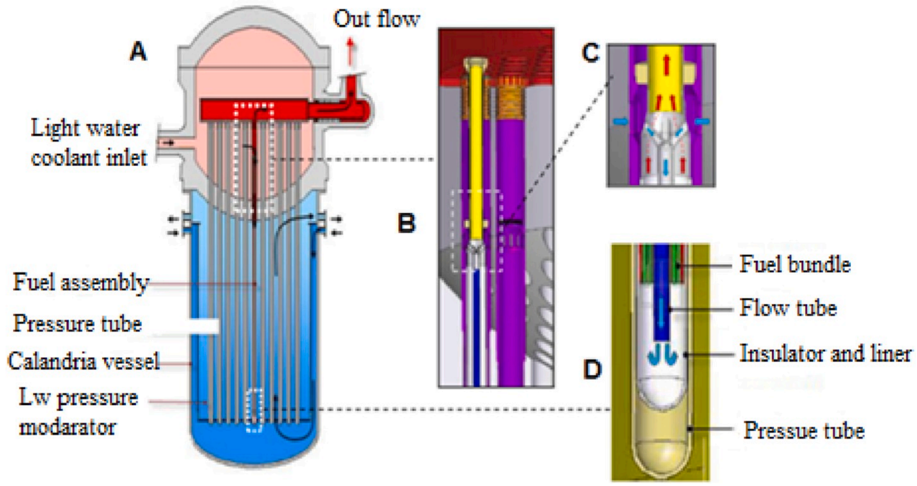

Fig. 19. Cross-sectional side view of Supersafe© reactor core and flow streams. A-Core layout, B- Pressure tube connection to tube sheet, C-Coolant flow from inlet plenum and flow to outlet plenum, D- Redirection of coolant flow (Courtesy of Yetisir et al., 2012).

Table 6
Comparison of Supersafe© fuel cycle parameters for the reference (Pu-Th) fuel cycle, LEU-Th, and LEU fuel cycle choices (Yetisir et al., 2012).

Characteristics	Reference (13% PUO_2/ThO_2)	LEU-Th (50 wt% LEU in Th)	LEU
Wt% of heavy elements (Initial fissile)	8.6% (Pu-239+Pu-241)	6.55 (U-235)	6%(U-235)
Exit burnup (Average) (MWd/kg)	44.3	42.9	44.2
Consumption of initial fissile (MWd/kg initial fissile)	515.1	681.5	738.3
Length of cycle (EFPD)	660	670	720

3. Nuclear submarine and merchant ship reactor design review

3.1. Introduction to nuclear submarine

3.1.1. Conventional submarines and air-independent submarine

Conventional submarines cannot use the combustion engine during underwater operation for the lack of oxygen. Large capacity batteries are used during the marine operation. In addition, these batteries needed to recharge by diesel engines during the above water navigation. Diesel-electric is a prevalent propulsion solution to non-nuclear submarines around the world (Wikipedia, 2018a). Fig. 20 shows the block diagram of a conventional submarine propulsion system. In order to prolong the marine operation time, the air-independent propulsion (AIP) design was proposed by some countries who wish to use non-nuclear submarines, such as Germany, Japan, France (Defencyclopedia, 2016). The essential of AIP is to produce oxygen by submarine itself. There are two methods to produce large amounts of oxygen; one is the High Test Peroxide (HTP), which is concentrated hydrogen peroxide at 85–98 percent (Wikipedia, 2018b). The peroxide can decompose into water and oxygen strongly with manganese as a catalyst. The other method is using stored liquid oxygen (LOX) from cryogenic tanks. Once oxygen is enough, the power can be generated by kinds of engines. The most preferred designs are the Stirling engine and fuel cell (Wikipedia, 2018c).

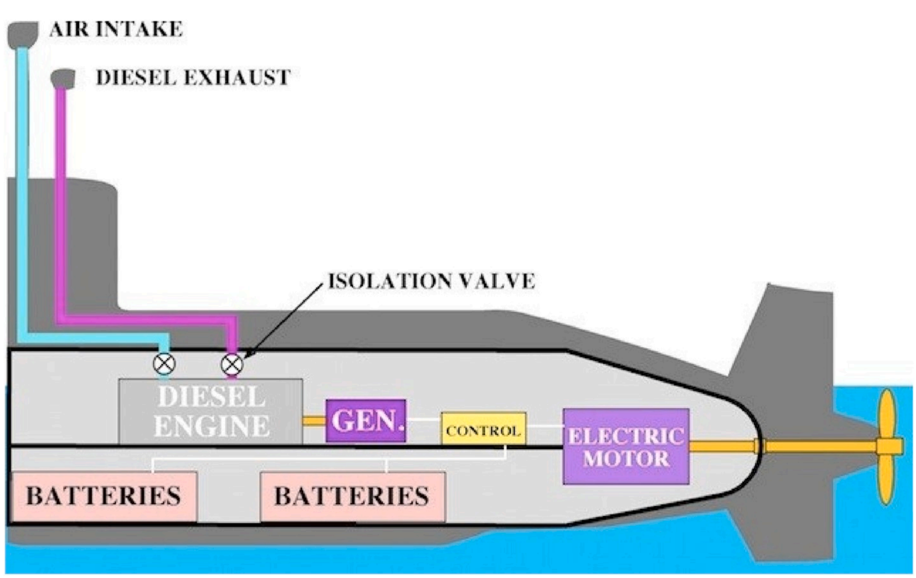

Fig. 20. Block diagram of a conventional submarine propulsion system.

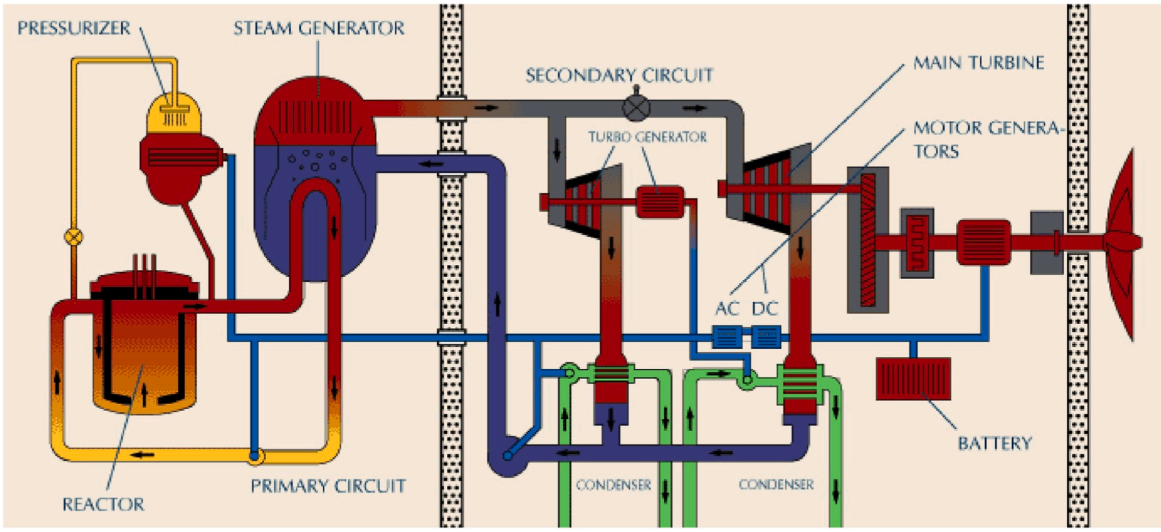

Fig. 21. Nuclear submarine propulsion system.

Table 7
Comparison of conventional submarine and nuclear submarine.

Characteristics	Types			
	Collins Class Australia	Type 218SG (derivative of 216 Singapore)	Virginia Class US	Astute class UK
Propulsion type	Diesel-electric	Fuel cell and lithium-ion battery	S9G Nuclear reactor	Rolls-Royce PWR 2 reactor
Propulsion power	5400 kW		30,000 kW	
Cost	$0.95 billion $110 million per unit (annual operating cost)	$1.09 billion $100 million per unit (annual operating cost)	$2.71billion $50 million per unit (annual operating cost)	$1.23 billion $50 million per unit (annual operating cost)
Displacement	3353 tons	4000 tons	7900 tons	7400 tons
Length	77.5 m	90 m	115 m	97 m
Beam:	7.8 m	8.1 m	10 m	11.3 m
Endurance	70 days	80 days (AIP endurance 4 weeks)	unlimited except by food supplies	unlimited except by food supplies
Speed	20knots	20knots	30-35 knots	30knots
Dive depth	180 m	300 m	240 m	300 m
Crew	58	74	134	109

3.1.2. Nuclear power submarines

The ultimate air-independent propulsion is a nuclear-powered steam turbine. For nuclear-powered submarines, the need for atmospheric oxygen can be eliminated since the steam turbine get heat from nuclear reaction rather than combustion, and nuclear reaction does not need any air. Another advantage of a nuclear submarine is that the breathing air can be recycled or produced by electrolysis, and freshwater can be distilled from seawater. Thus the endurance life was limited only by its food stores (Hirdaris et al., 2014a).

The advantage of a nuclear submarine is not only the inherent air-independent but also the high power volume density, which gives submarine a higher speed. A nuclear submarine also can be designed with a considerable energy capacity for an unlimited navigation range.

Fig. 21 is a typical nuclear submarine propulsion system. The main difference between the conventional submarine propulsion system and nuclear propulsion system is the prime mover, one is a diesel engine, and the other is a nuclear-powered steam turbine.

Table 7 is a comparison of three conventional submarines. The Virginia class information was obtained elsewhere (Wikipedia, 2018d). The nuclear submarine has twice a capital cost as traditional submarines. But the maintenance cost is half. As the designed life of a submarine is typically more than 25 years, which means nearly one billion dollars' maintenance cost can be saved for the use of a nuclear submarine. The nuclear submarine is more cost-competitive.

3.2. Nuclear submarine reactor review

Six countries have nuclear powered naval vessels: the US, Russia, the UK, France, China, and India. The US, France, and the UK had given up the conventional diesel-electrical submarines to develop and operate only nuclear submarines. More than 400 nuclear submarines were constructed by the end of the cold war in 1989, and 250 of which have been decommissioned under the weapons reduction agreement between east and west. Russia and the United States are the big two nuclear submarine holding countries with more than 100 each. The U.K. and France hold no more than 20 each. China may have 6 to 10 nuclear submarines. There is only one submarine in India (Magill et al., 2011).

3.2.1. American submarine reactor review

The US has developed 27 kinds of nuclear reactor designs using more than two hundred nuclear ships (Murray, 2009). Table 8 shows the US naval reactor designations convention: A three-letter name indicating

Table 8
The naval reactor designations in the US (Hirdaris et al., 2014a).

Ship types	Generation number	Contracted designers:
"A" – aircraft carrier	1	"B" – Bechtel
"C" – cruiser	2	"C" – Combustion Engineering
"D" – destroyer	3	"G" – General Electric(GE)
"S" – submarine	4	"W" – Westinghouse

the reactor application, generation, and designer, respectively.

For instance, an S8G reactor means GE's 8th generation submarine reactor.

3.2.1.1. First nuclear submarine reactor S1W. During the first nuclear submarine design stage in 1949, the two-reactor concept was proposed by General Electric (G.E.) Company and Westinghouse Company. G.E. introduced the submarine intermediate reactor (SIR) concept, which is more promising at first but given up later for a safety problem. The Westinghouse Company proposed the submarine thermal reactor (STR), which is essentially a pressurized water thermal reactor concept. Plutonium fuel had been considered first but given up later. In the end, the pressurized water thermal reactor named STR with the fuel of highly enriched uranium-235 proved to be most attractive. This reactor is the first nuclear submarine "Nautilus," designated as S1W, a PWR with highly enriched uranium fuel (Murray, 2009). The fuel elements in S1W are uranium-zirconium alloy plates. Fig. 22 is a dimensional drawing of the plate fuel elements with inch as a unit (Ragheb, 2011).

The second nuclear-powered submarine "Seawolf" in the U.S. initially had a reactor system that used liquid sodium as moderator and coolant. This system reduced the size of the reactor but was too difficult to maintain and was discarded after ship testing. The reactor was replaced by PWR later.

3.2.1.2. Innovative S5G prototype design. The S5G was innovative in the coolant circulation. It is designed in a natural coolant circulation, which means the reaction heat can be transferred by natural convection. In case of emergency, it still has motor pumps in the coolant circulation circuit; thus, it can also operate in forced coolant circulation. Natural circulation not only improves the safety for no safety issue, even loss of power, but also reduces the noise as the pumps are off in normal operation. The S5G maintained to run until the Navy's nuclear training program was decreased after the end of the Cold War. The natural circulation is achieved by using the stack effect or chimney effect (see Fig. 23).

The air temperature in the chimney bottom is higher than outside. Thus the density of the air column in the chimney is lower than outside. The pressure difference in the vertical direction will be different between outside and inside the chimney. The result is a pressure difference at the chimney bottom. This pressure maintains airflow.

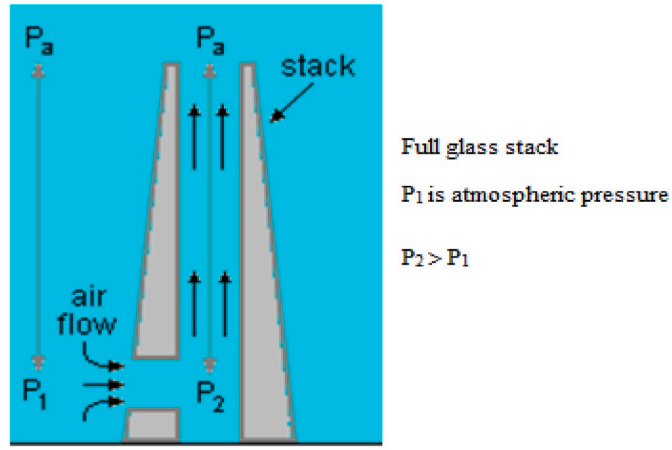

Fig. 23. The stack effect in chimneys.

The S5G design puts the reactor, which is the heat source on the bottom of the hull and the steam generators higher than the reactor. The reactor vessel is similar to a chimney; the natural circulation design was a success. The S5G is the quietest reactor in the US naval reactors with the rated output thermal power is 90 MW, the fuel is 93% enrichment U-235. The submarine noise comes from pumps, gearboxes, and propeller. To reduce the noise, S5G eliminates the pump's noise in normal operation. The successive design S8G also adopted the natural circulation design. This natural circulation design also reduces operational complexity and space. The S8G reactor with rated 35000 shaft horsepower (26.1 MW) was designed for the Ohio Class submarines. To improve the stealthiness, the S8G design adopts a giant propulsion turbine to eliminate the reduction gear.

3.2.1.3. Currently used S9G reactor. The S9G with 40,000 shaft horsepower (30 MW) is a PWR design used in Virginia Class SSN-774 submarines. This reactor has 33 years' design life without refueling. There is no refueling issue in this design; thus, the life cycle cost will be reduced. The S9G designs with a three-dimensional neutronics, thermal hydraulics, and structural mechanics-coupling model with the use of powerful

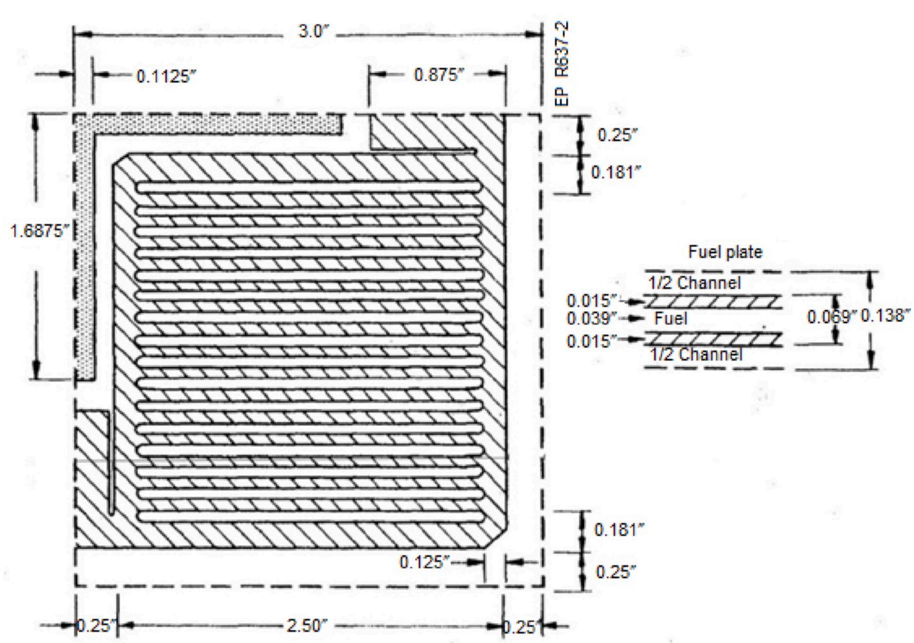

Fig. 22. Plate fuel element (Courtesy of Ragheb, 2011).

Table 9
Russian nuclear submarine reactor, icebreakers and battlecruisers (Wikipedia, 2018f).

Reactor system	Type	Thermal Power (MW)	First deploy	Purposes
OK-150	PWR	3*90	1959	Icebreaker *Lenin* before renovation
	PWR	2*70	1959	*November* class attack submarines *Echo* class cruise missile submarines *Hotel* class SLBM submarines
	LMFR	2*73	1963	Attack submarine K-27
OK-300	PWR	2*72	1967	*Victor I; II* class attack submarines
OK-350	PWR	1*89	1967	*Charlie* class cruise missile submarines
OK-700	PWR	2*90	1967	*Yankee* class SLBM submarines
	PWR	2*177	1969	*Papa* class attack submarine
OK-900	PWR	2*159	1970	Icebreaker *Lenin* after renovation
OK-550	LMFR	1*155	1971	*Alfa* class attack submarines
OK-700	PWR	2*90	1972	*Delta I* class SLBM submarines *Delta II* class SLBM submarines
OK-900A	PWR	2*171	1975	*Arktika*-class icebreakers
OK-700A	PWR	2*90	1976	*Delta III* class SLBM submarines
OK-300A	PWR	2*72	1977	*Victor III* class attack submarines
OK-650M	PWR	2*190	1980	*Oscar* class cruise missile submarines
OK-900	PWR	2*300	1980	*Kirov*-class battle cruisers
OK-650	PWR	2*190	1981	*Typhoon* class SLBM submarines
OK-650B-3	PWR	1*190	1983	Attack submarine K-278
OK-650A	PWR	1*190	1984	*Sierra* class attack submarines
OK-9VM	PWR	1*190	1984	*Akula*-class submarine
OK-700A	PWR	2*90	1984	*Delta IV* class SLBM submarines
KLT-40M	PWR	1*171	1988	*Taymyr*-class icebreakers
KLT-40	PWR	1*135	1988	*Merchant ship Sevmorput*
OK-900	PWR	2*171	1989	*SSV-33 Ural*-command ship
OK-650B	PWR	2*190	2009	*Borei* class SLBM submarine
OK-650V	PWR	1*200	2011	*Graney* class attack submarines
	PWR		2016	*LK-60Ya class icebreaker*
	PWR	2*150	2018	*Russian floating nuclear power station Akademik Lomonosov*

Computer-Aided Design (CAD). The S9G higher power density reactor results in a smaller, providing flexible arrangement. The S9G adopts a new propulsion system-pump jet propulsion rather than a traditional blade propeller. The new technology reduces the risks of cavitation, thus reduces noise caused by the propeller.

3.2.2. Soviet naval reactors

Both pressurized water reactors (PWR) and liquid metal fast reactors (LMFRs) are used in Russian nuclear submarines, but PWR is dominant (Wikipedia, 2018a). The Russian nuclear company OKBM Afrikantov who has no competitors, has designed 460 nuclear reactors and provided to the Russian navy. Table 9 represents the Russian nuclear submarine reactor, icebreakers, and battlecruisers.

3.2.3. UK naval reactor design

In the early days of the UK nuclear navy, the HMS Dreadnought nuclear submarines adopted the S5W reactors from US Company Westinghouse. Even the so-called first UK designed reactor-PWR1 still designed the core based on S5W. After 1958, the UK company Rolls-Royce designed the reactor by itself with some transferred technology from the US under the US-UK Mutual Defense Agreement (Wikipedia, 2018a). The Rolls Royce's second-generation reactor design-PWR2 is the commission reactor in submarines of the UK. The latest PWR2 does not need refueling in the whole submarine life, typically 25 years. There are some safety problems in PWR2 design, which leads to the need for next-generation design urgently (Wikipedia, 2018a). PWR3 is the name of the UK's next-generation reactor design. The new model once again based on a US design but will be manufactured by UK technology. The UK Ministry of Defense had selected PWR3 for successor design in May 2011.

3.2.4. French naval reactor

Although France's nuclear power industry is very thriving and advanced in the world, its nuclear submarine reactor designs have a prolonged development. The Rubis class is 1st generation nuclear submarine launched in 1979, and the Barracuda class is the 2nd generation nuclear submarine now commissioning. The designs are very compact. 1st generation reactor is a pressurized water K48 nuclear reactor (48 MW thermal power) with a refueling time of 7 years, 2nd generation reactor is pressurized water K15 reactor (150 MW thermal power) with a refueling time of 10 years.

3.2.5. Comparison of land-based nuclear reactor and nuclear submarine

The most significant difference between the land-based nuclear reactor and the submarine reactor is the operating condition. The land-based nuclear power plant provided the utility baseload, which is almost constant for some days. Thus, the design focuses more on stability. For the submarine reactor, the emphasis is the power operation flexibility since the operating condition is very complex and volatile during the war. Thus, the submarine reactor's thermal efficiency is lower than the power plant reactors. Some other particular concerns for submarine reactor design are noise reduction and space constraints. A nuclear submarine can use the vast seawater as a heat sink, while land-based reactors need an abundant water source and a massive cooling tower.

Reactors for submarines must be as small and powerful as possible. Therefore, it will have a very high power volume density, which suffers the components of the system. Its safety criteria are much stricter than land-based reactors for people in submarines are more vulnerable. Its system's reliability must be higher for its changing and rigorous operating conditions. Other issues are salt corrosion and sensitive controllability.

The fuel in the submarine reactor is a metal-zirconium alloy. The alloy is typically 15% Uranium with 93% Uranium-235 enrichment. If uranium percentage is higher, the lower enrichment is needed. The fuel in power plant reactors is usually UO_2 ceramic with no more than 5% enrichment. The higher U-235 enrichment, the longer the refueling period (see Table 10).

Nuclear power plants and earlier submarine reactors use the loop type coolant circulation. Modern submarine reactor designs adopt the integral circuit type (Ragheb, 2011). The steam generator has been integrated into the RPV for safety and structural compactness.

During the refueling of nuclear reactors and a nuclear submarine, spent fuel assemblies are unloaded before the loading of fresh fuel. The refueling necessitates the withdrawal of the upper part of the reactor pressure vessel. As the upper plenum of the core structure of SCWR is

Table 10
Comparison of the reactor in submarine and power plant.

Characteristics	Nuclear Power Plant	Submarine Propulsion system
Output power (MWe)	300MW–1600MW	30MW–165MW
Fuel	UO_2 ceramic	metal-zirconium alloy
Thermal efficiency	35%	33%
Fuel enrichment	(natural)0.72%–7%	20%–96% U-235
Refueling Period (year)	1–1.5	10–33
Space Size	Big	small
Power density	Low	Very High

simple, the refueling of SCWR is comparatively easier. Moreover, Moghrabi and Novog (2018) showed that the change in reactivity and the characteristics of four factors is quite similar in the case of fresh fuel and spent fuel for SCWR. However, a little difference is observed that caused mainly by the presence of Xenon, U-233, and variations in the concentration of isotopes Pu. Due to the possibility of application of on-line refueling in SCWR, the management of small change of reactivity is quite simple. Since most of the submarines are PWR type and upper plenum of the pressure vessel is comprised of all the control rods, several actions must be carried out to remain the core subcritical in the absence of control rods. Even at the End of Life (EOL) core, the excess reactivity is measured to be 3.5–5.0% Dk (Sarkisov and Tournyol, 1999). The estimation is performed to evacuate total water from the core of the reactor to maintain the core sub-critical. Nevertheless, the removal of decay heat from the spent fuel may become an issue. Moreover, after removing water, the radiation level will increase, and the appropriate radiation protection measures of workers have to be implemented (Takano et al., 2012).

3.3. Nuclear merchant ships

Civil merchant nuclear-powered ships have not expanded except a small number of experimental nuclear ships. In 1962, NS Savannah, The US build, a demonstration of civilian nuclear power, was completed. It was a tiny merchant ship with high-priced to run cost-effectively. The design was a highly compromised one, being neither a feasible passenger liner nor an efficient freighter.

The Otto Hahn, a cargo ship and research facility, developed by German, some 650,000 nautical miles (1,200,000 km) navigated on 126 voyages more than ten years period with no technical troubles. Nevertheless, it ascertained very expensively to run and transformed into diesel. The Japanese vessel named The Mutsu was insisted on technical and political difficulties. Its reactor had a radiation leakage significantly opposed by the fisherman to operate the ship. Low-enriched uranium fuel used in all of these three ships. Since 1988, Soviet LASH carrier with an icebreaker, Sevmorput, has been sailed on the Northern Sea Route successfully. It is the only merchant ship in service until 2012, powered by nuclear energy.

Civil nuclear ships bear dedicated infrastructure expenses. Initial costs of The Savannah were huge for the first civilian nuclear ship, including nuclear shore staff and servicing facility costs. Due to making only one ship, this infrastructure was an expensive one. A bigger nuclear fleet with using the same infrastructure would be able to reduce the consecutive increase in expenses: every ship would be low-priced than the previous one.

Currently, interest has been gowned in developing nuclear propulsion, and some draft proposals have already been made. In the case of a nuclear cargo ship, the cargo coaster (Wikipedia, 2018f; Jacobs, 2007), for example, is a new design. Japan has also been developed as an advanced integral type marine reactor called MRX (Marine Reactor X) (Kusunoki et al., 2007) (see Fig. 24). The reactor is made extraordinarily compact and lightweight by implementing water-filled containment. MRX is a PWR type reactor with a thermal output of 100 MW. The total weight of MRX is about 1600 tons, and its volume is 1210 m^3, which is nearly half than that of the first Japanese nuclear ship 'Mutsu'.

On the contrary, the power of MRX is higher three times than that of 'Mutsu'. The number of components (active) is decreased in the reactor than the loop type PWRs. Enhanced safety and reliability have also been achieved, which was evaluated by both experiments and analysis. Fig. 25 illustrates a cross-section of the pressure vessel (PV) and containment vessel (CV) of an MRX reactor. The core, steam generator (SG) and CRDM with pressurizer are located in the lower, middle, and upper part of the reactor pressure vessel, respectively. The direct connection of the primary coolant pumps is made to the vessel's flange.

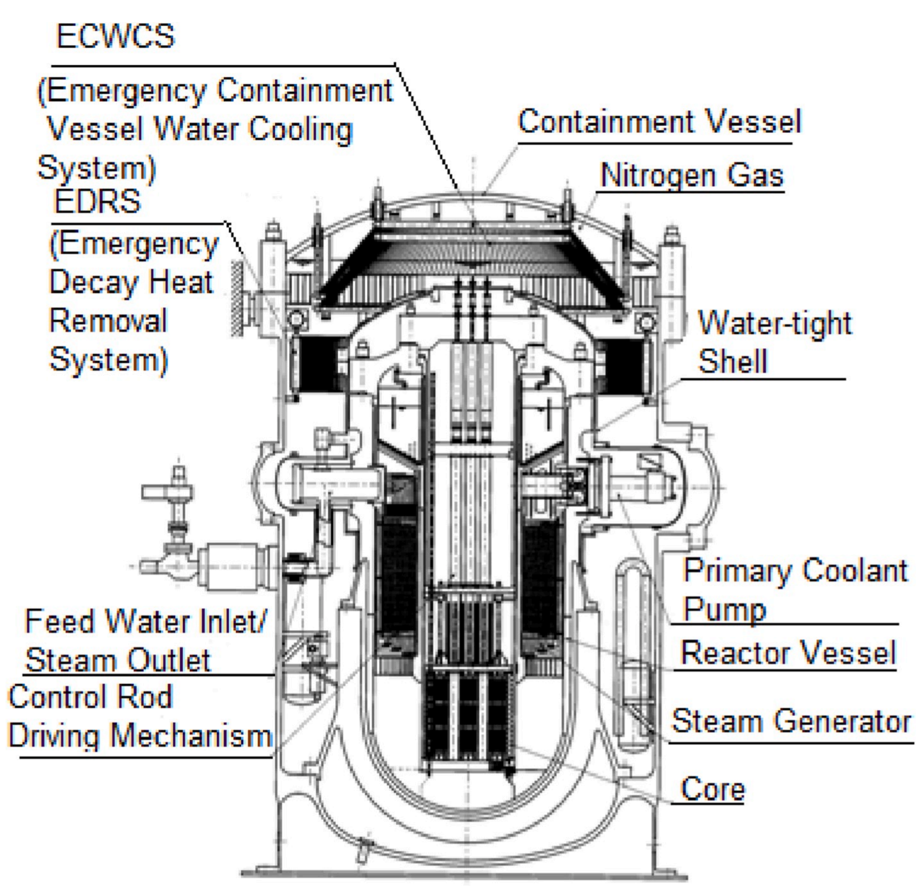

Fig. 24. Conceptual drawing of MRX. (Courtesy of Kusunoki et al., 2007).

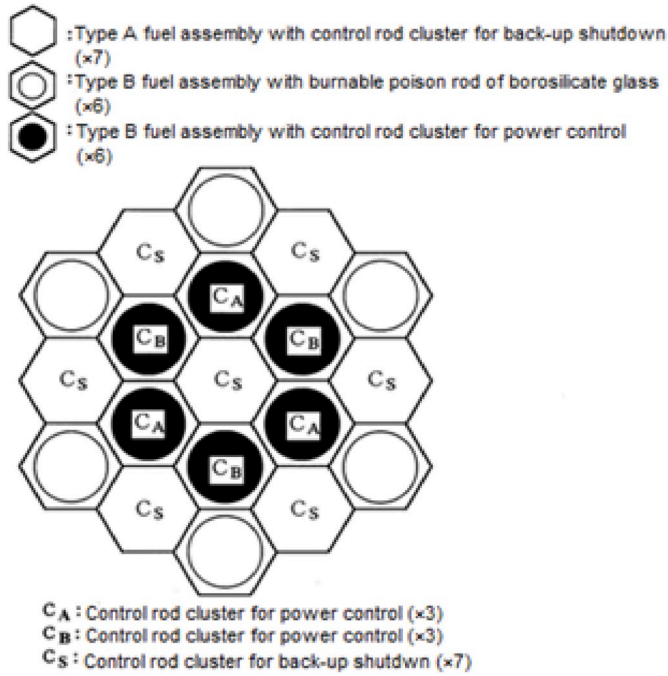

Fig. 25. Arrangement of fuel assemblies and control rod clusters (Courtesy of Hirdaris et al., 2014b).

Table 11
Major parameters of the Integral Type MRX reactor (Kusunoki et al., 2007).

Reactor power	100 MWt
Operating pressure	12 MPa
Inlet/outlet temperature of coolant	282.5/297.5 °C
Coolant flow rate	4500 t h^{-1}
Equivalent diameter of the core	1.49 m
Effective height of the core	1.4 m
Linear heat flux (average)	7.9 kW m^{-1}
Type of Fuel	Zry-clad UO_2 fuel
Enrichment of fuel	4.3%
Inventory of Fuel	6.3 ton
Burn-up of fuel (average)	22.6 GWd t^{-1}
Number of fuel assembly	19
Outer diameter of Fuel rod	9.5 mm
Control rod drive mechanism (CRDM) Type	In-vessel
Number of CRDM	13
Main coolant pump	Horizontal axial flow, canned motor type
Rated power	200 kW
Number of pumps	2
Type of steam generator	Once-through helical coil type
Material of tube	Incoloy 800
Outer/inner diameter of steam generator tube	19/14.8 mm
Steam pressure/temperature of steam generator	4.0 MPa/289 °C
Flow rate of steam	168 ton h^{-1}
Heat transfer area of steam generator	754 m^2
Inner diameter/height of reactor vessel	3.7/9.7 m
Type of containment	Water-filled RV immersion type
Inner diameter/height of containment	7.7/13 m
Design pressure of containment	4.0 MPa

The specifications of the principal parameters of MRX are shown in Table 11.

Fig. 25 illustrates the fuel assemblies and the control rod clusters. The MRX core comprises of 19 fuel-assemblies. The standard fuel-assembly includes the fuel rods (regular), the fuel rods together with Gd_2O_3, the glass rods of boron–silicate, and the control rod insertion pipe. There are three types of fuel assemblies named type A (seven assemblies), type B (six assemblies), and type C (six assemblies). Enrichments of the standard fuel rods and the rods with Gd_2O_3 are 4.3% and 2.5%, respectively. As cladding materials, the Zircaloy-4 is used while the B_4C is employed as control rods (cluster type) absorber. The power control of the MRX is made only with the control rods, and no chemical shim control is applied for the power control. This measure is taken to avoid re-criticality because of entering seawater into the core for the ship sinking. Table 12 presents the key parameters of the MRX core.

British Maritime Technology and Lloyd's Register, UK, Gen4 Energy (former Hyperion Power Generation), USA and the Greek ship operator Enterprises Shipping and Trading SA embarked upon a two-year study in November 2010 based on a 70 MWt reactor like Hyperion's to develop a concept of a tanker-ship design. Furthermore, Lloyd's Register updated the rules of the nuclear ship related to the incorporation of a reactor with the rest of the vessel certified by a land-based regulator. As a complete basis of the rule-making process, it is presumed that the nuclear regulators, in the future, will make sure that the nuclear plant operators will establish safety in operation. It will replace the present practice of the marine industry where the builder usually creates regulatory compliance, in addition to safety through design and construction. At present, owner countries are responsible for their nuclear ships, and no one is engaged in global commerce. In 2014, Lloyd's Register along with other associates of the consortium presented an initial design concept study of Suezmax tanker for a load of 155,000 dwt baizing on a standard hull form with different arrangements for containing a 70 MWt plant of nuclear propulsion releasing up to shaft power of 23.5 MW at highest steady rating (average: 9.75 MW) (Hirdaris et al., 2014a, 2014b). It is a fast-neutron reactor of small size employing cooling using lead-bismuth eutectic and capable of running before refueling for ten full-power years, which remain in service functioning life of the vessel for 25-years. They concluded that the idea is practicable, but additional development of nuclear technology and the progress and standardization of the regulatory structure would be required beforehand.

4. SCWR in ship application

SCWR is the only reactor design using water as coolant among the six reactor types that are being investigated in the GEN-IV international advanced reactor development program. As SCWR uses water as a coolant, it is very convenient to get sufficient water from the operating environment of ships, as ships operates in the water. More importantly, the SCWR system is very compact (steam generator, steam separator, steam dryer, recirculation pump, etc. is not required), which is suitable for transportation applications. Moreover, the enthalpy content of the coolant is high, which eventually reduces the coolant mass flow rate per unit core thermal power. The result is the reduction of the reactor coolant pumps, piping, condenser, and related equipment size, and the pumping power. These features will ultimately reduce capital and operating costs. Of course, these cost savings cannot be determined until a more detailed design of a supercritical marine plant is formulated. Due to the significant reduction of size, weight, capital, and operating costs of SCWR, there is a vast potential of application of SCWR in ship and submarine.

Considering the performance requirements for nuclear submarine reactors, the Japanese SCWR design and the European HPLWR design could be more attractive. All existing nuclear ships and submarines are pressure vessel type, and both the Japanese and European SCWR designs are also based on pressure vessel type reactor, which is, therefore, preferable for ship applications. The design parameters of these SCWR reactors need to be scaled down for ship application, and the verification and validation of the design necessitates further experiments and quantitative analysis. The Canadian Small Modular SCWR is of pressure tube type design; consequently, investigations on suitability of pressure tube type reactor for ship applications are required.

However, the Canadian Small Modular SCWR is a scaled-down version of Canadian SCWR. It is, therefore, finds application in land-

Table 12
Key parameters of the MRX core (Kusunoki et al., 2007).

Parameters	Condition	Value	Design conditions
K-eff	BOC (cold shutdown, all control rod clusters are inserted)	0.82962	
	EOC (hot and full power, withdrawn of all control rod clusters)	1.02041	
Reactivity coefficient	BOC		
	Doppler coefficient	$-2.2 \times 10^{-5}\Delta k\ k^{-1}/^{o}C$	
	Void coefficient	$-2.5 \times 10^{-3}\Delta k\ k^{-1}$/% void	
	Moderator density coefficient	$3.0 \times 10^{-1}\Delta k\ k^{-1}/(g\ cm^{-3})$	
	EOC		
	Doppler coefficient	$-2.3 \times 10^{-5}\Delta k\ k^{-1}/^{o}C$	
	Void coefficient	$-2.6 \times 10^{-3}\Delta k\ k^{-1}$/% void	
	Moderator density coefficient	$3.2 \times 10^{-1}\ \Delta k\ k^{-1}/(g\ cm^{-3})$	
Reactivity shutdown margin		2.17% $\Delta k\ k^{-1}$	≥1.0%
Heat flux		1.427 kW m^{-2}	
Maximum linear heat rate		30.4 kW m^{-1}	<41 kW m^{-1}
Fuel center temperature (full power, 1200 MWD/t)		1785 °C	
Minimum DNB (100% power)		2.25	≥1.73

based electricity generation where low energy demand (due to say a sparsely distributed population) does not warrant the use of a full-scale SCWR. Until now, several Japanese studies showed that the Japanese SCWR is the most matured technologies in terms of safety during transient and accident conditions (Okano et al., 1994; Oka et al., 2010; Ishiwatari et al., 2006, 2005; Cai et al., 2009). However, none of the SCWR reactor concepts has proven safety by demonstration. Statistically, the nuclear propulsion system has a negligible influence on the safety of the whole naval unit. Reactor related accidents caused only a small number of casualties.

Based on the particular requirements on nuclear submarine reactors, overall design parameters of SCWR in the marine application is proposed. The proposed design parameters are as follows: follows: Inlet temperature 270–280 °C, Outlet temperature 500–530 °C, PV pressure 25 MPa, 67–367 MW thermal power with an efficiency of 45% corresponding to the 30–165 MW electricity power. This design uses light water as both coolant and moderator. The suggested design is mainly a preliminary conceptual design based on some existing designs. The proposed design needs to evaluate quantitatively.

There is a possibility of applying the supercritical nuclear power plant technology for driving marine propulsion as an alternative for PWR presently used in submarines. It is expected that the SCWR system retain many of the advantages that have made pressurized-water plants successful in marine applications, not the least of which is the inherent load-following capability. Due to the improvement of the thermodynamic cycle in supercritical design, a significant increase in plant efficiency is therefore expected. In addition, this SCWR technology offers a substantial saving in size and weight. These features supposed to reflect on the reduction of operation and capital costs. However, the actual extent of this cost-saving cannot determine until a detailed design of the supercritical marine plant is framed. The overall justification behind the proposal of using SCWR in marine includes (i) lower steam mass flow rates, which eventually reduce dimensions of the turbo set and the condenser, and (ii) considerably higher efficiency, of about 41.8%, which is much higher in comparison of 32.1% of the two-contour system. It is expected that increased efficiency affects the cost of the naval application.

Calculations and preliminary design of SCWR for use in marine have been worked out in some countries (Piwowarski, 2014). An analysis of supercritical turbine propulsion system was done in the Gdansk University of Technology, Faculty of Ocean Engineering and Ship Technology Narutowicza, Poland. They proposed a supercritical reactor cooled with a light water system for the naval application. In the evaluated SCWR system, the steam generated in the reactor reaches 26 MPa pressure and 570 °C temperature, internal turbines efficiencies were assumed of 90%, condenser pressure was 6 kPa. The power output of the turbine thought to 70 MW, gross efficiency of this cycle was 41.8%, and the mass flow rate was 44.43 kg/s; turbine revolutions were 6000 rev/min. Preliminary thermodynamic and flow calculations made the basis for designing turbine flow parts.

The supercritical reactor plant represents an advance in technology, and it is noteworthy that modern technology makes it possible to use more advanced safety systems of this type in submarines, due to the used materials and safety of workers. Further investigations require the corrosion behavior of clad and heat shield materials and the nature and extent of radioactive corrosion product carryover into the turbine. Therefore, a supercritical plant could be designed and constructed when a satisfactory solution to the material problems is achieved.

5. Safety analysis of SCWR

The safety analysis is performed to ensure the safety response of the plant during regular operation, transient and accidental situations. In the case of regular operation, the behavior of SCWR is different from other water-cooled reactors due to the absence of a recirculation loop similar to BWR and primary circuits similar to PWR. As the supercritical water passes directly from the RPV to the turbine system, hence no steam-water separator and steam dryer are required. Thus, safety features should be different in the case of SCWR. Safety analysis can be divided into two parts, deterministic safety analysis, and probabilistic safety analysis. They must ensure that all barriers are capable of holding the integrity of the reactor, and no (or minimal) radioactive materials will be released to the environment. During the design process, safety analysis is vital and has been made, considering the designs and approaches of different reactors. Yamaji et al. (2006) suggest that during abnormal transient situations, the safety attempt should be assured the maximum cladding surface temperature (MCST) to lower than 850 °C. On the contrary, in accidental conditions, core damage may occur, and Okano et al. (1997) predicted that MCST should be less than 1260 °C to avoid serious environmental consequences. The safety analysis should incorporate the design of fuel assembly, emergency core cooling systems, and response to transient and accident situations. However, the core is simple in design and several passages, supply natural circulation to enhance safety. Nevertheless, there are three safety points of view. The number of passes through the core is a more straightforward design and provides natural circulation that improves safety, however, the following three concerns appear (i) large axial picking factor, (ii) at the end of the fuel life cycle; it is hard to guarantee a negative void reactivity coefficient, and (iii) adjacent to the channel, a substantial increase in enthalpy may cause the fuel elements to overheat. These problems can be solved in either or both the ways: (i) by applying fuel of different enrichment in the axial direction, (ii) extra moderator at the core's upper part. The most notable change in SCWRs is the fuel cladding material; the widely utilized zircalloy does not retain necessary strength at supercritical water conditions. Therefore, it must be substituted with different materials.

At present, all over the world, several SCWR designs are under development. Moreover, due to its compact size makes it feasible to be applied for marine applications like nuclear propulsion system or floating nuclear power plant. However, the development must be conceived by ensuring the safety of operation for both normal and accidental situations. Rowinski et al. (2018) presented a review of the safety analysis of SCWRs considering different reactor designs and approaches available in the literature. They found that among all the SCWR designs, Japanese designs developed by Tokyo University and Waseda University are the most progressed in terms of the whole system designs. In terms of safety, they performed complete analysis during operation, transient, and accidental situations for both super LWR and super FR. The core damage frequency is under the needed maximum. At present, the European SCWR is still under the design phase, where safety analysis is being performed only in normal operating situations. However, several test facilities are under development to perform experimental works. The Canadian SCWR designs were examined for ensuring the essential average outlet temperature, and in case of regular operation, maximum cladding surface temperature lies within the safety margins. Numerical analysis was performed to optimize the fuel assembly.

Zhang et al. (2014) revealed that in case of safety analysis of SCWR, the loss of coolant accident (LOCA), specially the depressurization process, is one of the challenges. LOCA is the most attractive design basis accident in the safety analysis of SCWR. Unlike the PWR and BWR, the LOCA for SCWR has faster depressurization and further remarkable transitions from initial supercritical pressure. Due to a substantial property change near the pseudo-critical temperature of SCWR, accidental depressurization will sternly endanger rector core safety (Licht et al., 2008). In different thermodynamic regions, supercritical blowdown meets the difficulties of trans-critical pressure drop and break discharge rate. Consequently, the blowdown from initial subcritical situations is more complex than that of initial supercritical conditions. Considering the complexities, Zhang et al. (2014) presented models and developed a program to simulated supercritical blowdown from simple vessel. Finally, they investigated in detail the blowdown of supercritical water from simple vessel.

6. Conclusions

Supercritical-Water-Cooled Reactors (SCWRs) are considering worldwide over the existing light water reactor (LWRs) because of several potential benefits. The SCWRs offer some advantages in terms of high thermal efficiency without considering the boiling temperature limit, considerable plant simplification with free of the recirculation system, steam-water separators and steam dryers of BWRs as well as steam generators of PWRs. One of the main objectives of the development of SCWR concepts is to meet-up the difficulties of the market economy by cutting down capital cost utilizing the experiences of LWR and technologies of the supercritical coal-fired power plant. SCWR is only a GEN-IV reactor design using water as a coolant. Different countries chose different SCWR designs based on their experience. A review has been made of different SCWR designs, with particular focus on the fuel and core designs of Japanese, European, and Canadian reactors.

Besides, in this study, a review of existing nuclear power ship technologies of the US, UK, French, and Russia has been made. Besides, a comparative analysis of nuclear power ship with land-based SCWR is also conducted. Moreover, the development of Japanese integral type marine reactor (MRX) and civil nuclear merchant ships have been discussions. The supercritical design retains many of the advantages of existing LWR and supercritical coal-fired power plant and offers significant savings in weight and size and a substantial increase in plant efficiency. These features should reflect themselves in reduced operating and capital costs. The size of these cost savings can only be evaluated after formulating a further detail design of an SCWR for the marine vessel.

The present analysis reveals that there is a possibility to introduce SCWR in marine vessels due to several advantages of SCWR technology over existing LWR plants in-terms of size, weight, costs, efficiency, etc. Considering these benefits, a preliminary design of SCWR for a marine application is proposed following Japanese and European SCWR designs. Both Japanese and European SCWR is a pressure vessel type, which is similar to all existing naval nuclear vessels. Only Canadian SCWR and Small Modular Supersafe© reactor is the pressure tube type design. European HPLWR developed based on the Japanese SCWR. Considering all this information, Japanese and European SCWR may more suitable to consider for the ship application with the necessary scale down of the design parameters.

A supercritical reactor plant represents an advancement in technology. However, some issues should be further investigated as e.g., the production of supercritical steam in a radiation field that requires further investigations into the corrosion behavior of clad and heat shield materials and the nature and extent of radioactive corrosion product carryover into the turbine. In this respect, a supercritical plant could be designed and constructed when a satisfactory solution to the material problems is obtained.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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