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Science and Technology of Supercritical Water Cooled Reactors: Review and Status

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Abstract

An overview of the global development of Supercritical Water-Cooled Reactors (SCWRs) has been given. The SCWR concept is the natural design path and ultimate evolution of conventional reactors today as the vast majority of modern power nuclear reactors are water cooled units. The move from subcritical to supercritical pressures by the thermal power industry over the past 50 years has been successful. The SCWR concept follows two main types of large reactor vessel as found in conventional Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR); and distributed pressure tubes as found in Canadian and Russian nuclear reactor designs. The concept has been designed to operate with a thermal spectrum core, fast spectrum core or mixed spectrum core. SCWRs promise to increase the efficiency of modern nuclear power plants from 30 - 35% to about 40 - 45% and reduce capital and operational costs. Most of the SCWR designs are still at the conceptual stage and are not expected to be implemented till 2030 because of the challenges faced in thermal-hydraulics and construction materials chemistry that would withstand supercritical conditions.

Keywords: SCWR, PWR, BWR, thermal spectrum, fast spectrum, mixed spectrum, capital and operational costs.

1. Introduction and Historical Development of SCWRs

As a clean energy source nuclear energy has become very important on a global scale, undergoing rapid development which developing countries such as Nigeria are now encouraged to harness for base-load units within their energy matrix. Nuclear Energy holds the potential to provide a clean, reliable and affordable supply of energy for meeting the growing needs of Nigeria's economy while protecting the environment and ensuring energy security. From the global nuclear park today, water-cooled reactors are the dominant product design of nuclear power plants and moving from subcritical to supercritical conditions has been the natural design path in the evolution of water cooled reactors over the past 50years (Pioro and Duffey, 2007). However, the current generation of water cooled reactors has some shortcomings from long-term development point of view ranging from economic competitiveness to low fuel utilization. In order to eliminate these shortcomings, some developed countries like Japan, China, Canada, and Russia have started R&D work on nuclear power plants with supercritical water-cooled reactors (SCWR) concept.

According to Pioro and Duffey (2007), the design of SCWRs is seen as the natural and ultimate evolution from today's conventional nuclear reactors obtaining its main features from the modern PWR at high pressure of ~16MPa, direct-cycle or once-through design of BWRs, steam superheaters from experimental reactors, and modern supercritical turbines, at pressure of ~25MPa and inlet temperature of ~600^0C that has been operating successfully at thermal plants for many years. Therefore, supercritical Nuclear Power Plants (NPPs) have higher operating parameters when compared to modern conventional NPPs as shown in Fig 1.

An SCWR is a direct cycle nuclear system that operates under supercritical pressure conditions (~25 MPa). The coolant at the outlet of the reactor core has a temperature higher than 500^oC and goes directly to the turbine. With a thermal efficiency as high as 45%, the SCWR has much higher thermal efficiency than existing water-cooled reactors (Oka, 2000). In addition, nuclear power plants using supercritical water as coolant have no need of steam generator, pressurizer or steam separator. Hence, the cooling system is significantly simplified as shown in Fig. 1.

Also, direct thermo-chemical or indirect electrolysis production of hydrogen at low cost could be possible because of the higher supercritical water temperatures. The low cost is due to increased process efficiencies. According to IAEA (1999), the optimum required temperature is about 850^oC and the minimum required temperature is around 650^oC to 700^oC, well within modern material capability.

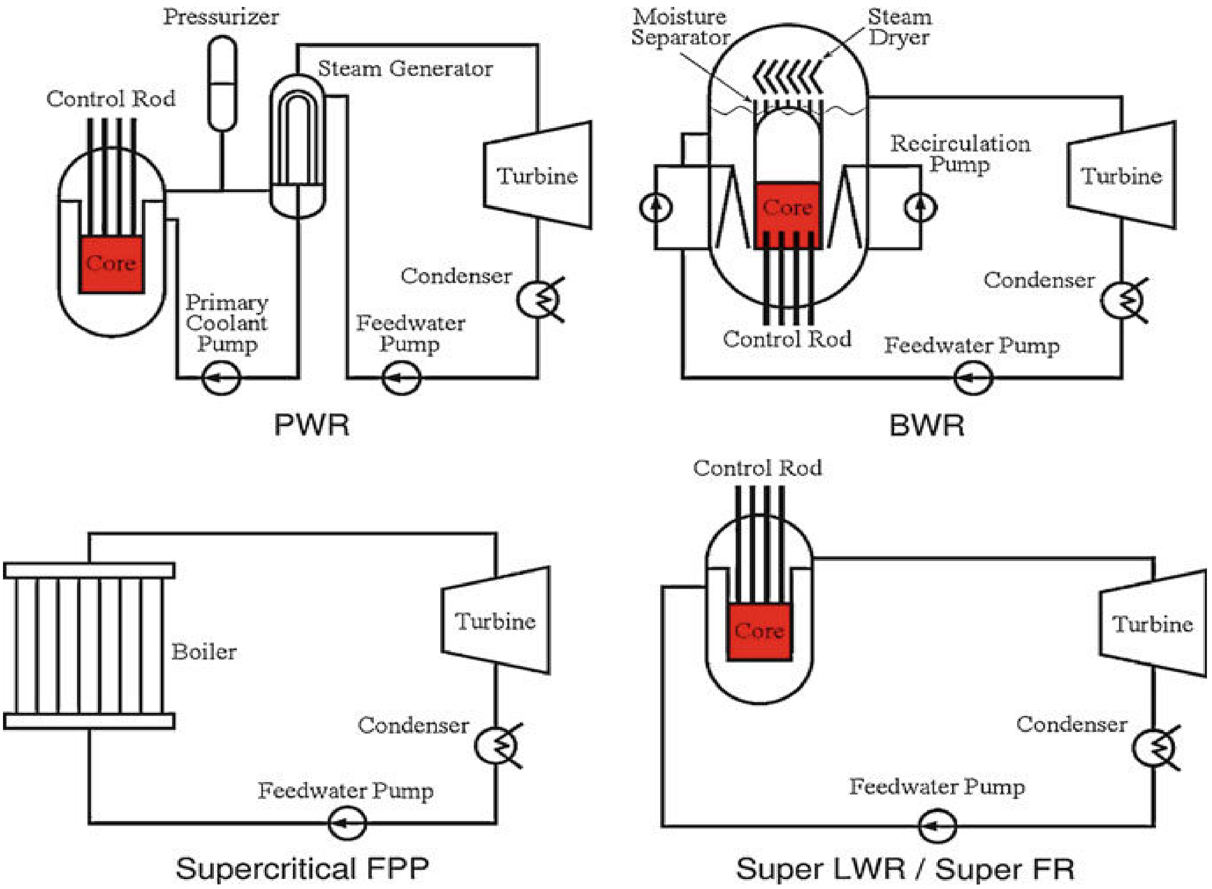

Fig. 1: Schematic Diagram of Plant Systems (Oka et al, 2010)

SCWR concepts have been studied as early as the 1950s and 1960s in the USA and former USSR. The table below lists the major characteristics of the first concepts of SCWRs.

Table 1: First concepts of SCWR (Oka, 2002; 2000)

	Company/Reactor Acronym (year)			
	Westinghouse		**GE, Hanford**	**B & W**
Parameters	**SCR(1957)**	**SCOTT-R(1962)**	**SCR (1959)**	**SCFBR (1967)**
Reactor Type	Thermal	Thermal	Thermal	fast
Pressure, MPa	27.6	24.1	37.9	25.3
Power, MW (thermal/electrical)	70/21.2	2300/1010	300/~	2326/980
Thermal Efficiency, %	30.3	43.5	~40	42.2
Coolant Temperature, at the outlet, °C	538	566	621	538
Primary Coolant flow rate, kg/s	195	979	850	538
Core height/diameter, m/m	1.52/1.06	6.1/9.0	3.97/4.58	~
Fuel material	UO_2	UO_2	UO_2	MOX
Cladding material	SS	SS	Inconel –X	SS
Rod Diameter/Pitch, mm/mm	7.62/8.38	~	~	~
Moderator	H_2O	Graphite	D_2O	~

Explanations to the table:
Acronyms: GE – General Electric; B & W – Babcock & Wilcox; SCR – SuperCritical Reactor; SCOTT-R – SuperCritical Once-Through Tube reactor; and SCFBR – SuperCritical Fast Breeder Reactor

2. Requirement and Consideration of SCWRs

An SCWR, from a technological perspective, is an integration of already existing generations of water cooled reactor technology and supercritical fossil-fired power generation technology. These already existing technologies ensure technological availability for the development of SCWRs.

The main objectives of using supercritical water in nuclear reactors are to (Pioro and Duffey, 2007):

1. Increase the efficiency of modern nuclear power plants (NPPs) from 33%-35% to about 40%-45%; and
2. Decrease capital and operational costs and hence decrease electrical energy costs.

Increased efficiency arises primarily from higher outlet temperatures and cost reduction is expected from the simplification of the design with the absence of components such as steam separators and steam dryers; smaller condenser and reactor building; and a reduction in the number of steam lines.

The SCWR concept is classified as a Generation IV reactor, where the Generation IV project, initiated by the United States Department of Energy's (US DOE) Office of Nuclear Energy, Science and Technology, is a new generation of nuclear energy systems that can be made available to the market by 2030 or earlier, and that offers significant advances toward challenging goals.

In a broad sense, requirements and considerations which could be used for the design of SCWRs are given below according to (Aksan, 2011):

- **Generation IV requirements**: The design of SCWR takes into consideration the goals of generation IV reactors which can be categorized into 4 broad areas viz: sustainability (fuel utilization and waste management), safety and reliability, economics (competitive life cycle and energy production cost), and proliferation resistance/physical protection.
- **European Utility Requirements (EUR):** This was developed by the major European utilities to define the features of future plants one of which is SCWR. The EUR scope was to allow development of competitive, standardized designs that would match the conditions in Europe and be licensable in the respective countries (EUR, 2010).
- **IAEA Safety Requirements:** IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) has established a set of requirements, organized in a hierarchy of basic principles, user requirements and criteria, comprising an indicator and an acceptance limit in all areas that must be fulfilled by an innovative nuclear energy system (INS) such as SCWR to meet the overall target of sustainable energy supply.
- **Western European Nuclear Regulator's Association (WENRA):** The design of SCWR also follows the guidelines of WENRA which is a network of Chief Regulators of EU countries with nuclear power plants and Switzerland as well as of other interested European countries which have been granted observer status. The main objectives of WENRA are to develop a common approach to nuclear safety, to provide an independent capability to examine nuclear safety in applicant countries and to be a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues.

3. Global Development of Scwr Concept

Thirty years after the concepts of nuclear reactors cooled with water at supercritical pressure were studied in early 1950s and 1960s in the USA and former USSR (Oka, 2000), the idea of developing a Supercritical Water-Cooled Reactor became enticing as the ultimate development path for water cooled reactors. A number of countries (Canada, Germnay, Japan, Korea, Russia, USA and others) have active R&D programmes on this concept and the following Table 2 shows the design parameters for these countries' SCWR concepts for easy comparison. However, this manuscript reviews the global development of SCWR concept in Canada and European Union with emphasis on thermodynamic cycle, core design, fuel design and safety systems.

Table 2: Design Parameters for SCWR concepts (Leung, 2011)

Parameters	Units	Canadian SCWR	SCWR-M	HPLWR	JSCWR	Superfast Reactor	SCWR-SM	VVER-SCP	US SCWR
Country	-	Canada	China	EU	Japan		Korea	Russia	USA
Organization	-	AECL	SJTU	EU-JRC	Toshiba /U. of Tokyo	U. of Tokyo	KAERI	OKB “Gidropress”, IPPE	INEEL
Reactor Type	-	PT	RPV	RPV	RPV	RPV	RPV	RPV	RPV
Spectrum	-	Thermal	Mixed	Thermal	Thermal	Fast	Thermal	Fast-resonance	Thermal
Power Thermal	MW	2540	3800	2300	4039	1602	3182	3830	3575
Linear max/aver	kW/m		39/18	35/14, 8, 4.5 (a)	-/13.5		39/14.26	-/15.6	39/19.2
Thermal eff	%	48	~44	43.5	42.7	~44		43-45	45
Pressure	MPa	25	25	25	25	25	25	24.5	25
T_{in} Coolant	^0C	350	280	280	290	280	280	290	280
T_{out} Coolant	^0C	625	510	500	510	508	510	540	500
Flow Rate	Kg/s	1320	1927	1179	2105	820		1890	1843
Active core height	m	5.0	4.5	4.2	4.2	2	3.66	4.05	4.27
Equiv. core diameter	m	~4.55	3.4	3.8	3.34	1.86		3.6	3.93
Fuel	-	Pu-Th	UO_2/ MOX	UO_2	UO_2	MOX	UO_2	MOX	UO_2
Cladding material	-	SS	SS	316SS	310SS	SS		Austenitic alloy (ChS-68, EP-172)	SS
No of FA	-	336	284	1404	372	162/73	193	241	145
No of FR in FA	-	78	180/324	40	192	252/127	316	252	300
D_{rod}	mm	7/12.4/12.4 (b)	8	8	7	5.5	9.516	10.7	10.2
Pitch	mm	Vary	9.6/9.6	9.44		6.55	11.5	12	11.2
moderator	-	D_2O	H_2O/--	H_2O	H_2O	-/ZrH	ZrH_2	H_2O	H_2O

Explanation of Table:

(a)Evaporator, superheater 1, superheater 2 (b) Outer, middle and inner rings

3.1 Canada’s SCWR Concept

In this concept, the main features of the Canadian Deuterium Uranium (CANDU) reactor (Modular fuel channels and Heavy water moderator) are retained in addition to supercritical light water coolant at pressure of 25MPa and outlet temperature up to 625^0C. Besides, separation between moderator and coolant is unique to CANDU reactor which is an advantage in the area of enhanced passive safety system especially when the moderator (Heavy Water) acts as a passive heat sink.

Fuel Design

- Three concentric rings of fuel with 15, 21 and 42 fuel element with fuel composition being 13% plutonium in thorium
- A large non-fuel element in the center (Zirconia surrounded by cladding) which reduces coolant void temperature.
- Fuel cladding option could either be austenitic stainless steel, martensitic steel or oxide dispersion strengthened steel.

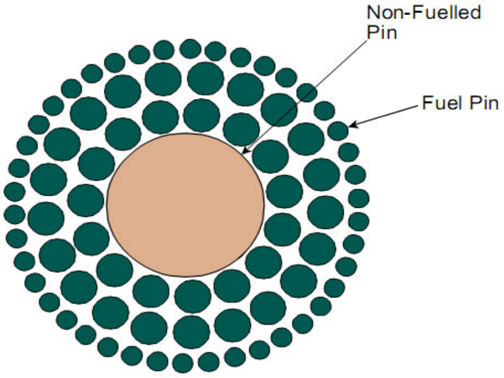

Fig 2: Fuel Design (Yestisir, 2011)

Core Design

- Vertical channels
- High pressure inlet plenum which simplifies refuelling process, reduce lattice pitch and relatively low temperature (350^{0}C)
- Low Pressure moderator
- Channel outlets connecting to header with small diameter reducing material thickness requirement

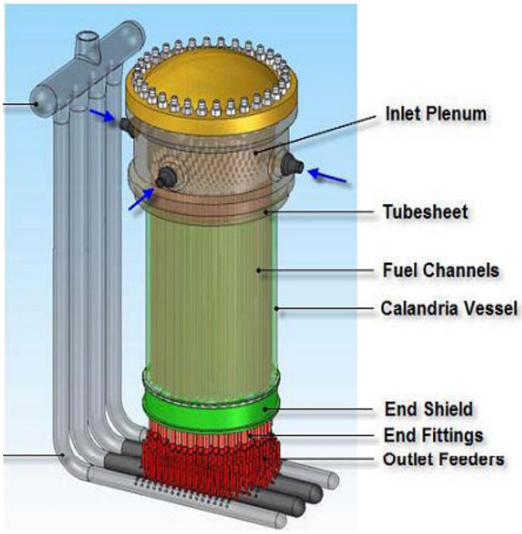

Fig 3: Core Design (Yestisir, 2011)

Thermodynamic Cycle

- Similar to the current advanced turbine configuration of Supercritical fossil power plant
- High-pressure "steam" is fed directly into the steam turbine(direct steam cycle) which ensures improved efficiency and plant simplification
- Moisture separator reheater reduces the steam moisture inside the low pressure turbines

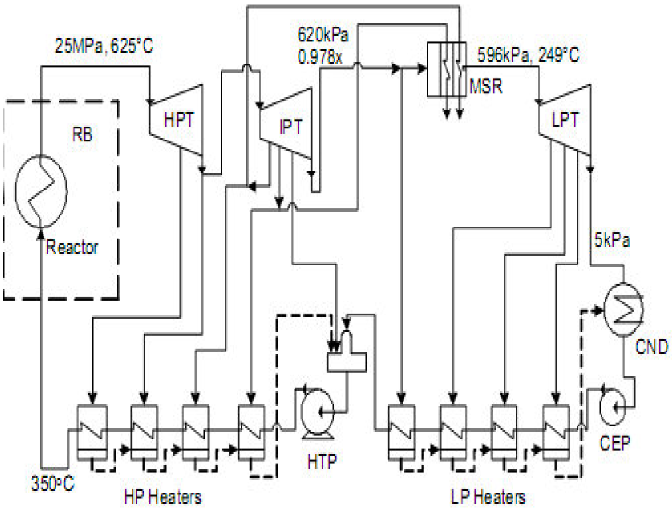

Fig 4: Thermodynmica Cycle (Yestisir, 2011)

3.2 European Union's SCWR Concept

The European Union's SCWR concept is named High Performance Light Water Reactor (HPLWR) which is of reactor pressure vessel type with thermal power of 2300MW and electric power of 1000MW. It operates at 25MPa pressure with core exit temperature of 500^0C. HPLWR has a three-zone core: evaporator, superheater 1 and superheater 2 as shown below.

Fuel Design

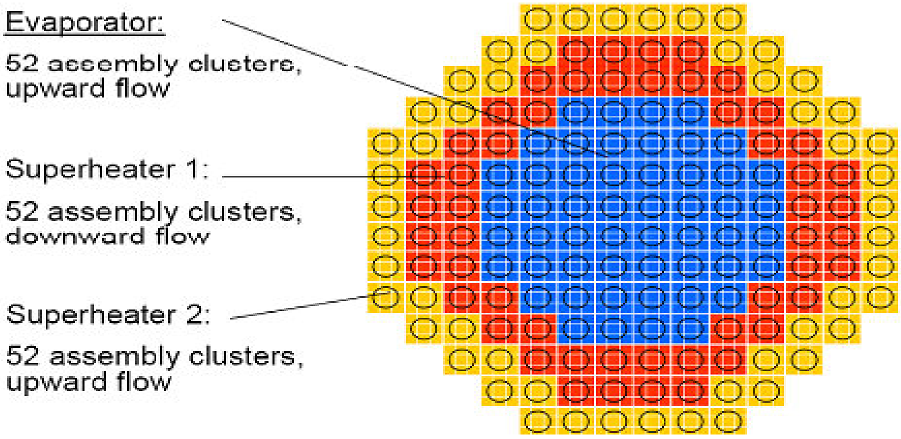

Fig 5: HPLWR three-zone core (Schulenberg, 2009)

Core Design

- Equivalent core diameter is 3.8m
- Inlet flow splits into two:
  - Upward to cool the dome and down through the gap between assemblies
  - Downward to lower plenum at temperature from 280^{0}C to 310^{0}C
- Heat-up is in 3 steps with coolant mixing between steps to eliminate hot streaks:
  - Upflow in evaporator (310^{0}C to 390^{0}C)
  - Downflow in superheater 1 (390^{0}C to 433^{0}C)
  - Upflow in superheater 2 (433^{0}C to 500^{0}C)

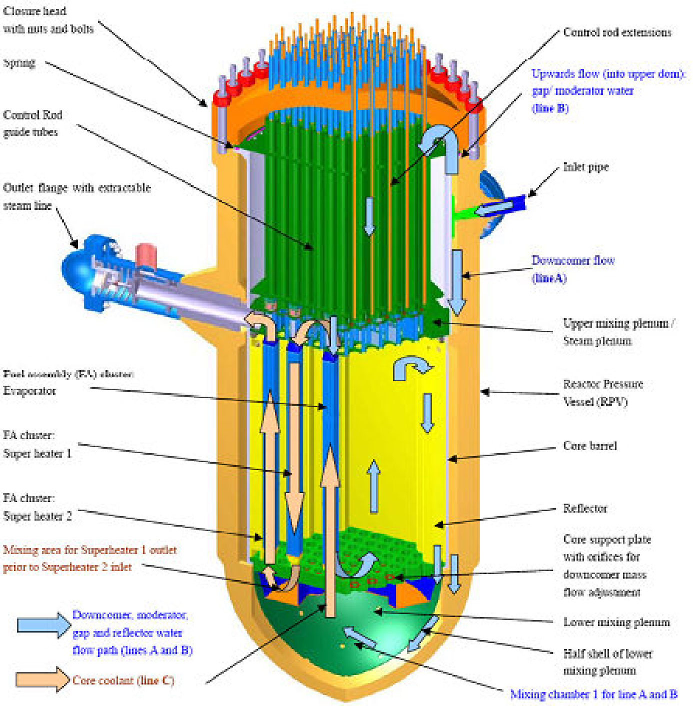

Fig. 6: HPLWR Core Design (Schulenberg, 2009)

Thermodynamic Cycle

- It uses BWR concept and supercritical pressure fossil-fired power plant
- Steam from the reactor is fed to the high pressure turbine, undergoes re-heat using part of the extracted steam and then enters the intermediate pressure turbine and low pressure turbine.

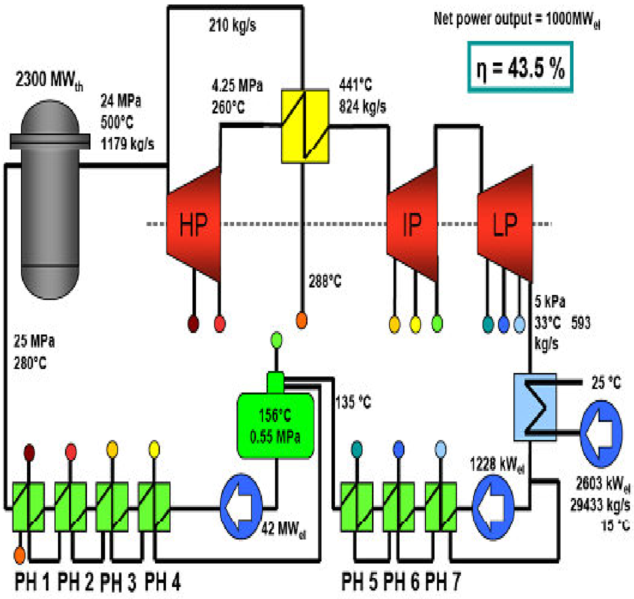

Fig 7: Thermodynamic Cycle (Brandauer et al , 2009)

4. SCWR Safety

Existing SCWR designs are in conceptual or preconceptual design stages, thereby making safety analysis difficult as there is a lack of operational experience and reduced levels of specific information, and that goes without saying that such design stages are the most cost effective time to improve safety of design. The effectiveness of a Probabilistic Risk Analysis (PRA) which provides a systematic method for achieving safety goals may be limited by information availability early in the design phase but it is easier to make corrections earlier in the design phase.

Most international designs (Japan, EU, Canada) have the Advanced Boiling Water Rector (ABWR) safety systems as a practical starting point with similar shutdown system(s); Safety Relief Valve (SRV) for pressure relief; containment venting; active and passive Residual Heat Removal (RHR); Emergency Core Cooling (ECC) systems based on power availability; and use of passive systems, core catcher, containment cooling, and hydrogen mitigation in the case of severe accidents.

Some general issues for consideration in the safety system design in the case of the HPLWR (SCWR) include:

- the case of the water inventory within the primary system being ~1/8 compared to a BWR or PWR – thus it is mandatory for all sequences to maintain coolant flow through the core.
- the heat transport capacity in the case of loss of off-site power influenced by inertia of the pumps is provided for by a flywheel for main coolant pumps and sufficient inertia of main coolant pump and motor; for PWRs and BWRs respectively – thus such a measure should be decided for the SCWR

As SCWR neutronics does not uniquely require any new physics to be modelled, existing codes may be upgraded to an extent of being able to simulate transients. Though, implementation of adequate coupling of thermal hydraulics and neutronics; and experimental data is needed for validation of the codes. A model for heat transfer deterioration is also necessary. Code-by-code comparison for selected transients is also encouraged to validate integrity of results.

5. Challenges and On-Going Research & Development

Common SCWR design challenges affect four areas namely: Materials, Chemistry, thermal hydraulics and safety.

Materials: There is no single alloy with sufficient information to confirm its performance for in-core components (like internals and fuel cladding) of SCWR. There is need for research on materials that are thermal and corrosion resistant. However, for out-of-core components, materials could be selected based on the materials characteristics for Supercritical fossil-fired power plant.

Chemistry: The available experimental data showed that there is a rapid change in chemical properties the coolant due to change in Supercritical water (SCW) density near critical and pseudo-critical point which has strong impact on corrosion and stress corrosion cracking (SCC). Moreover, SCWR in-core radiolysis is markedly different from those of conventional water cooled reactors which makes extrapolation of behaviour inappropriate.

Thermal-hydraulics: Due to lack of phase change in SCWR, cladding temperature limits are now the design criteria in lieu of the traditional Critical Heat Flux (CHF) criteria. Then, accurate prediction of SCW heat transfer is essential to establish the power output and safety margin and this requires experimental data for relevant bundle geometry at conditions of interest that are not available. Most available experimental data on SCW heat transfer were obtained with tubes which are applicable to fossil-plant boiler but not directly to SCWR geometry and conditions.

Safety: There is need for establishment of design-basis-accidents and potential initiators of severe accidents which covers: need for transient experimental data on supercritical heat transfer, experimental SCW data on critical flow for the design of safety/relief valve and depressurization systems, experimental data and analytical model to predict the onset of instability in the system, coupled neutronics and thermal-hydraulics analysis which is required for design calculation and integral test data at supercritical conditions to validate outcome of safety analysis codes.

In the light of all the aforementioned challenges, the need for R&D arises mainly from the differences of the reactor systems in the areas of supercritical pressure water as coolant, high temperature/pressure in the core and the design of the plant total system. However, no significant R&D will be necessary for Balance of Plant (BOP) because SC fossil-fired plant technologies can be applied with the inclusion of radioactivity in the main steam line.

6. Conclusion

Global SCWR concepts have successfully been proposed over time as the basic science of SCWR neutronics generally has no unique (new) physics to be modelled; only the geometry, temperature and properties are different from the conventional. Though many reactor physics codes can perform calculations under these conditions, the validity and accuracy of these codes needs to be rigorously demonstrated (especially with issues of higher fuel and moderator temperatures and harder neutronic spectrums). Coupling thermal hydraulics and netronics is however required to reflect specific features of SCWR cores.

The established science and technology of supercritical fossil-fired power plants and water cooled reactors has provided a good resource base for development of SCWR concepts. However, challenges arise as reactor vessel integrity is of particular concern to SCWR cores and vessel materials in order to produce low leakage cores due to SCC and fatigue issues at weld overlays and coatings as some vessels may be vulnerable to embrittlement at supercritical conditions.

Further research is encouraged in the aforementioned areas and newcomer countries such as Nigeria should look towards advanced reactor technologies such as SCWRs with attractive advantages of improved efficiencies and reduced costs in order that upon deployment of such reactor system, it would be affordable after the resolution of all the challenges and safety issues.

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