ONCE-THROUGH CYCLE, SUPERCRITICAL-PRESSURE LIGHT WATER COOLED REACTOR CONCEPT

Yoshiaki OKA and Seiichi KOSHIZUKA
Nuclear Engineering Research Laboratory, The University of Tokyo
3-1, Hongo 7-Chome, Bunkyo-ku, Tokyo 113-8656, JAPAN

ABSTRACT
Concept of once-through cycle, supercritical-pressure light water cooled reactors was developed. The research covered major aspects of conceptual design such as cores of thermal and fast reactors, plant system and heat balance, safety system and criteria, accident and transient analysis, LOCA, PSA, plant control and start-up. The advantages of the reactor lie in the compactness of the plant from high specific enthalpy of supercritical water, the simplicity of the once-through cycle and the experiences of major component technologies which are based on supercritical fossil-fired power plants and LWRs. The operating temperatures of the major components are within the experience in spite of high coolant outlet temperature. The once-through cycle is compatible with the tight fuel lattice fast reactor because of high head pumps and small coolant flow rate.

Keywords: supercritical-pressure, light water, once-through cycle, concept design, nuclear reactor

1. INTRODUCTION

Innovation of nuclear power reactor is necessary to compete with advanced fossil-fired power plants in deregulated market. Innovation of fast reactor is also necessary to find a way of competitive plutonium utilization. We developed the concept of once-through direct-cycle supercritical-pressure, light water cooled reactors for the purposes [1]. The critical pressure of water is 22.1MPa. The supercritical water does not exhibit a change of phase. Heat is effectively removed at the pseudo critical temperature, 385C at 25MPa which corresponds to boiling point at subcritical pressure . The steam water separation is not necessary in the once-through cycle where whole coolant is sent to turbines.

Although several supercritical-pressure reactor concepts were reported in 1960's and 1990's [2], only the present concept takes the once-thorough cycle and light water cooling with a reactor pressure vessel (RPV). Here the water cooling means that the core inlet coolant is high density water, not steam above psuedocritical temperature. The plant system is compared with those of BWR, PWR and the supercritical fossil- fired power plant (FPP) in Fig.1.

The primary advantage of the reactor concept is the compactness of the reactor and the containment because of the high specific enthalpy of supercritical water. The present concept first showed the advantage among the supercritical-pressure reactor studies [3]. Simplicity of the once-through cycle plant system is the other advantage. The difference from other studies also lies in its wide scope. Almost all aspects of conceptual design have been studied by developing computer codes for the analysis. It includes not only the core design, but safety design, transient and accident analyses including loss of coolant accident (LOCA), probabilistic safety assessment (PSA), plant control and start-up, stability, plant heat balance and thermal efficiency.

The advantage in manufacturing and reliability of the major components results from the operating experience of the components in LWR and supercritical thermal power plants. The RPV and control rod drives are similar to those of PWR. Containment and emergency core cooling system (ECCS) are to those of BWR. The turbines, pumps and main steam pipings are to those of supercritical thermal power plants which adopted once-thorough cycle 40years ago.The operating temperature of the major components are within the experiences of those of LWR and supercritical thermal power plants in spite of the high outlet coolant temperature above 500C. The fuel cladding and fuel assembly are the major developmental component in relation with supercritical water chemistry, but they are exchangeable.

The supercritical pressure once-through cycle is also compatible with tight-fuel-lattice fast reactor core because of the high head pumps and small coolant flow rate, approximately one eighth of LWR. The advantage of high power density of fast reactor over thermal reactors will increase the competitiveness of the fast reactor over LWR. Breeding is possible in the fast reactor, although the breeding ratio is not so high as that of the liquid metal cooled fast reactors. This paper summarizes the result of the studies.

2. CORE DESIGN

The core can be designed both the thermal and the fast reactor. The thermal reactor is called SCLWR and the fast one is SCFR. The high temperature versions are called SCLWR-H and SCFR-H respectively. The outlet coolant density is less than one third of BWR. The moderation is provided by the water rods of the fuel assembly in the thermal reactor. Fast reactor adopts tight fuel lattice without water rods.

The core design criteria are;

1.maximum cladding temperature 450C for stainless steel (SS) and 620C for Ni-alloy cladding
2. maximum liner heat rating; 39kw/m
3. negative coolant void reactivity for both thermal and fast reactors

The maximum cladding temperature criterion is conservatively determined for avoiding oxidation corrosion of the claddings for the purpose of concept development. But the temperature limit must be experimentally verified in the future. There is no such criterion as the minimum critical heat flux ratio of LWR. This is one of the critical improvements in the concept development of the once-through cycle reactor. The coolant flow rate of the once-through reactor is inevitably small because of no recirculation of coolant. When taking the similar criterion as minimum critical heat flux ratio (MCHFR) of LWR for transients, the coolant flow rate need to be kept high and enthalpy rise in the core cannot be large. The heat transfer deterioration is not so violent phenomena as burnout. The cladding temperature does not increase sharply and deterioration disappears in the down-stream. For the purpose of evaluating the cladding temperature directly during transients when heat transfer deterioration occurs, it was necessary to develop the data base of heat transfer coefficients at various conditions of heat flux, flow rate and coolant enthalpies The data base was prepared by the numerical simulation using k-epsilon turbulent model which successfully analysed the deterioration phenomena.This made it possible to design the high temperature reactors; SCLWR-H and SCFR-H taking high enthalpy rise and low coolant flow rate which is the advantage of once thorough cycle. The supercritical water is single phase fluid. This is an advantage in computational analyses, compared with the two-phase flow of LWR. The operating pressure is 25MPa, which is close to that of supercritical fossil-fired power plants, 24.2MPa. The fuel cladding is thick enough not to be collapsed at the high pressure(27.5MPa). The fuel rods are internally pressurized.

3. THERMAL REACTOR

The coss section of a fuel assembly is depicted in Fig.2 [4]. Since the coolant density decreases substantially in the upper part of the core, many water rods are introduced in it. The water rod is surrounded by almost stagnant water which has an insulation cover around. The space between the rod and cover is filled with supercritical water and it is axially divided by partition plates every 2cm. The partition plates are effective to avoid natural convection. The water rods are thermally insulated from the hot coolant of the fuel channel and good moderation is provided. Various types of water rod concepts, such as single tube, double tube, semi-double tube and also zirconium hydride rods were studied [5]. The downward flow in water rods which was proposed by TEPCO study [6] is taken in the present design. In the core with descending flow in water rods, part of feedwater is fed to the upper dome of the reactor pressure vessel and flow down through the control rod guide tube and the water rod. It is mixed with the rest of feedwater from the downcomer in the lower plenum and flow up through the fuel channel. The advantages of the downward flow are avoiding the thermal fatigue of the control rod guide tube, high outlet temperature and good moderation in the upper part of the core. The demerit is the longer fuel assemblies. The fuel enrichment and gadolonia concentration are axially changed in three zones for flattening the axial power distribution. SRAC code system of JAERI is used for neutronic calculation, but all other computer codes were developed for the concept development of superciritical-pressure reactors. Both axial and radial power distribution are estimated by coupling neutronic and thermal hydraulic calculation. The local power distribution of each fuel rods are flattenened by taking three enrichment splits. The characteristics of the SCLWR-H is shown in Table1.

The control rod clusters are adopted for primary reactivity control system. The drive mechanisms are mounted on the top of the RPV as those of PWR. The secondary shutdown reactivity is provided by the borated water injection system as that of BWR. Both systems can shut the reactor down at cold condition. The control rods and the RPV are similar to those of PWR. All RPV walls are cooled by the inlet coolant as in PWR. The feedwater temperature of SCLWR-H is 280C. It is lower than that of PWR in spite of the high coolant temperature. This is the advantage of the reactor concept in the strength of RPV. Only outlet nozzles are exposed to the hot coolant. Thermal sleeves will be provided there. The conventional steel of the PWR vessel is used. The shell can be fabricated by forging in Japanese factory. The shell of 1570 MWe SCLWR-H is 36.5 cm thick and 860 ton in weight, when the same steel as that of LWR is used. It is lighter than that of 1350 MWe ABWR, 910ton. It was confirmed by the one-dimensional transport calculation that the fast neutron irradiation damage of the vessel wall is within the limit in 100years. The numbers of inlet-outlet coolant lines are only two and small in diameter in spite of the 1000MWe class electric output. The containment is compared with that of ABWR in Fig.3. The volume is substantially small because of the high specific enthalpy of supercritical water and simple once through cycle system.

4. FAST REACTOR

The fast reactor, SCFR and SCFR-H adopted tight fuel lattice without water rods. The spacing between fuel rods is 1.3mm. The plant system are common with the thermal reactor except that accumulators are necessary for emergency cooling at loss of coolant accident(LOCA). The supercritical once-through cycle is more compatible with tight lattice core than LWR from the small core coolant flow rate, pumping power and stability. The negative reactivity at coolant loss was achieved by inventing the zirconium-hydride layer concept, placing thin zirconium hydride layers between seeds and blankets [7]. Fast neutrons at voiding are moderated through the layer and absorbed in the blankets. The neutron balance of the reactor becomes negative at voiding. The cross section of the core and pressure vessel of SCFR-H is depicted in Fig.4. The white hexagon shows the driver (MOX) fuel assemblies, while black one does the blanket. A radially heterogeneous core is taken for calculating the core performance in two-dimensional R-Z model without homogenizing the zirconium hydride layers. Optimization of the core arrangement remains for the future study.

The characteristics of SCFR-H is shown in Table1. The core design criteria are the same as the thermal reactor. The kinetic energy of the coolant in the core is taken as similar value as that of liquid metal fast breeder reactors for avoiding flow-induced vibration of fuel rods. Descending flow cooling in blanket is taken in the design. The flow path is similar to that of descending flow in water rods of the thermal reactor. When the outer most layer of blankets of SCFR-H are replaced by the driver fuel assemblies, the reactor power increases to 2017MWe with decreasing the conversion ratio to 0.96 [8]. The breeding ratio of 1.02 was attained for the core with large fuel rods of 1.02cm in diameter [9]. The power density of SCFR-H is higher than that of SCLWR-H. It means that the fast reactor will be more economical than the thermal reactor when MOX fuel is available with reasonable cost. This has been the goal of fast reactor development for long time.

5. SAFETY

The fundamental safety requirement is "maintaining core flow". "Maintaining feedwater from the cold leg" and "keeping coolant outlet open at the hot leg" are the requirements for the safety of the supercritical water cooled reactor which has the once-through coolant system. Coolant flow rate at the inlet and outlet of RPV are monitored and used for the emergency signal, instead of the "water level" of a BWR. The plant and safety system are shown in Fig.5. Reactor scram, high pressure auxiliary feedwater system (AFS) and low pressure core injection system(LPCI) are actuated succeedingly when the mass flow rate decreases 90%, 20%, and 10% of the nominal one. The system pressure is maintained at the supercritical pressure by the turbine control valves within the pressure change below 0.8%. The turbine bypass valves will be actuated between 0.8 and 4% and safety relief valves (SRV) will be opened for the changes larger than 4% (1MPa). When the system pressure becomes too low, the coolant will be released through the automatic depressurization system (ADS).

The major safety criterion of the accidents is that the maximum stainless steel (SS) cladding temperature should be below1260C for avoiding core damage. It is the same as the USNRC criterion for the LWR with SS cladding. Major criterion for transients is that the fuel integrity should be maintained. This is described that the maximum cladding temperature should be below 610C for stainless steel cladding and 840C for Ni-alloy cladding. This criterion depends on the fuel rod design and cladding strength.

The large break loss of coolant accident (LOCA) were analysed by the computer code, SCRELA [10]. The fuel cladding temperature after the 100% cold leg break is shown in Fig.6. The peak temperature is below the criterion, 1260C. The peak temperature is higher in SCFBR due to the high power density. The temperature rise in reflood was consistent with that of high conversion PWR of similar coolant to fuel volume ratio. The core is abruptly flooded at the hot-leg break, but the power increases only 20% in SCLWR. It dose not impose a problem of fuel integrity. LOCA is the limiting accident in the design of SCLWR and SCFR as in PWR, while it is not in BWR. The sensitivity study showed that the peak cladding temperature decreases by increasing downcomer height.

The flow, pressure and reactivity induced abnormalities were analyzed by the computer code which was developed for the analysis [11]. Nine types of accidents and transients of SCFR and SCFR-H at the supercritical pressure were analyzed. They are summarized in Table 2 and Fig.7. All satisfied the criterion.

No natural circulation coolant path exists in the once-through cycle reactor when main feedwater pumps stop. The effect of this features on core damage frequency is assessed by simplified probabilistic safety assessment (PSA) method [12]. It is concluded that the CDF is not high. Although no natural circulation is established at total loss of feedwater flow in the once-through coolant system, the core damage frequency is maintained as the same level of Japanese conventional BWR because of the diversity of feedwater systems in the direct cycle reactors.

6. RESEARCH PROGRAMS

Academic research of supercritical water chemistry, irradiation damage and heat transfer deterioration at the University of Tokyo were funded for 4 years from JSPS (Japan Society of Promotion of Science) a subsidiary of Monbusho in 1998. European HPLWR, high performance light water reactor research program started in 2000 with the research fund of European Union. The authors are invited to participate in the program. The research and development of SCR are funded in 2001 from Japanese Ministry of Economy, Trade and Industry (METI) under the program of supporting innovative nuclear technologies at Institute of Applied Energy (IAE). The research team consists of the people from Toshiba, Hitachi, Kyushu University and the University of Tokyo and is lead by Toshiba. Plant conceptual design, thermal hydraulics, material and water chemistry will be studied in the 5 year program.

9. CONCLUSION

The concept of once-through cycle supercritical pressure light water cooled reactors was developed. It is based on the water coolant technology of the fossil-fired power plants and LWR. Fundamental safety principle of the reactor is maintaining" "coolant flow rate" instead of" "water level" of LWR. The safety design and criteria are similar to those of LWR. The reactor system is compact and simple because of the high specific enthalpy of supercritical water and the elimination of the recirculation and steam-water separation systems. Fundamental guidelines in designing the reactor was developed. Development of fuel cladding remains for the future study.

ACKNOWLEDGMENT

The authors are indebted to that people of Japanese manufacturers and Tokyo Electric Power Company (TEPCO) for valuable discussions and comments. The authors express sincere thanks to TEPCO for a part of financial support

of the study and to the people of Japan Atomic Energy Research Institute (JAERI) for making the SRAC code [13] available for the neutronic calculation.

REFERENCES

1. Oka,Y., and Koshizuka,S., Proc. SCR2000 Symposium, paper No.101 November 6-9, 2000, Tokyo
2. Oka,Y., Proc. SCR2000 Symposium, paper No. 104, November 6-9, 2000, Tokyo, The University of Tokyo
3. Oka,Y., et al., Proc. 10th Pacific Basin Nucl Conf. Kobe, October 20-25, 1996, vol.1, pp.779- 786, (1996).
4. Dobashi,K. et al., Ann.Nucl.Energy, vol.25, pp.487-505 (1998)
5. Okano,Y., Koshizuka,S. and Oka,Y., Ann.Nucl.Energy, vol.21, pp.601-611, (1994).
6. Tanaka,S. et al., Proc. ICONE-4, vol.2, pp.199-211, ASME (1996).
7. Oka,Y. and Jevremovic,T., Ann.Nucl.Energy, vol.23, pp.1105-1115 (1996)
8. Oka,Y. et al., Proc. ICONE 8, ICONE 8216, April 3-6, 2000, Baltimore ASME (2000)
9. Ishiwatari,Y., Oka,Y. and Koshizuka,S., Proc. of ICONE-9, paper No. 305, April 9-12, 2001,Nice, France, ASME.
10. Lee, J.H., Koshizuka,S. and Oka,Y., Ann.Nucl.Energy, vol.25, pp.1341-1361, (1998).
11 Kitoh,K., Koshizuka,S. and Oka,Y., Proc. ICONE-7, ICONE-7234, ASME, 1999.
12. Lee,J.H., Oka,Y. and Koshizuka,S., Reliability Engineering & System Safety, vol.64, pp.327-338, (1999)
13. Okamura, K.,Kaneko,K., Tsuchihashi, K.,"SRAC 95" JAERI-Data/ Code 96-015, (1996)

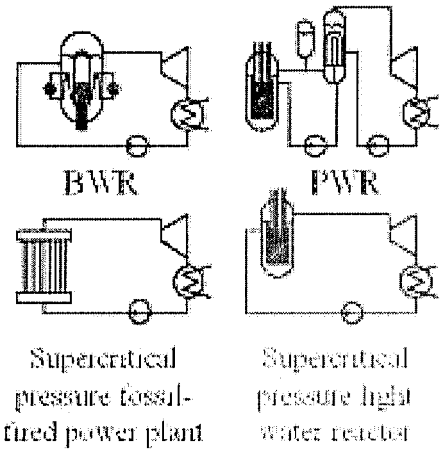


Fig.1 : Plant systems

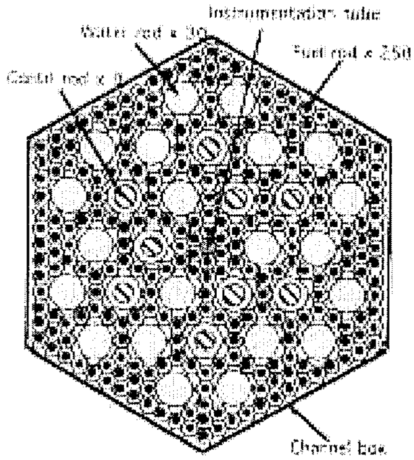


Fig.2 : Fuel assembly

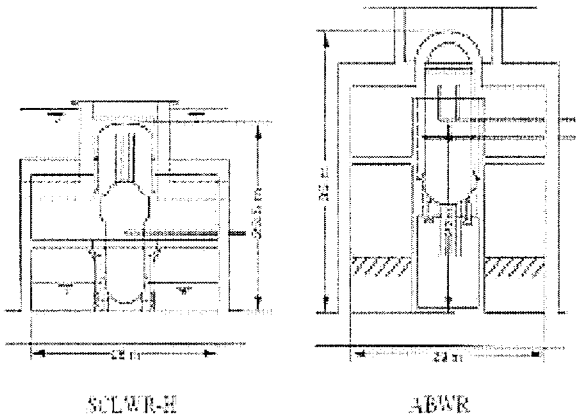


Fig.3 : Comparison of containments

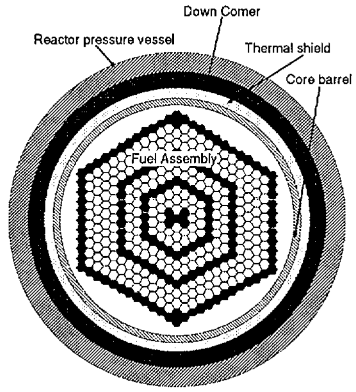


Fig.4 : Cross section of the core and pressure vessel of SCFR -H

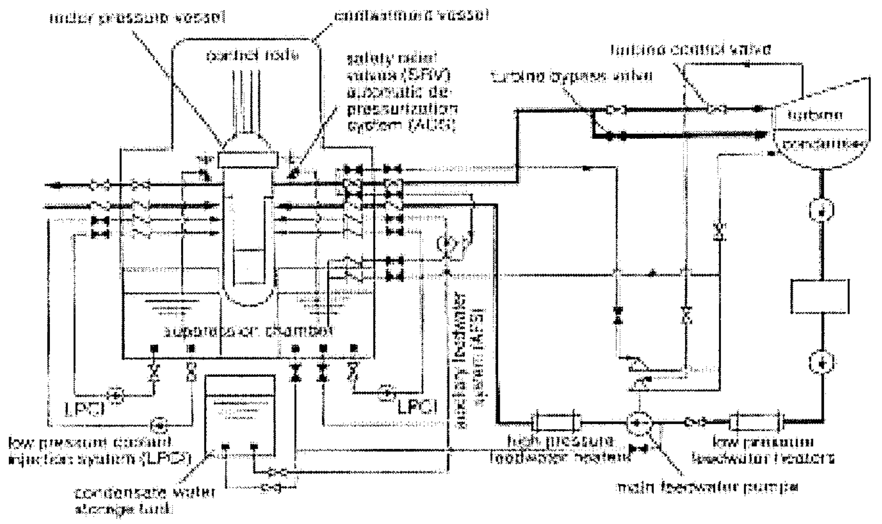


Fig.5 : SCLWR plant and safety system

Table.1 : Comparison of core characteristics of ABWR, SCLWR-H, and SCFR -H

Thermal/ electric power output(MW)	3926/1356	3586/1570	3893/1728
Pressure(MPa)	7.2	25.0	25.0
Thermal efficiency(%)	34.5	44.0	44.4
Cladding	Zr		
	ascending		
	872		
	3.71/5.16		
	50.6		
	278/287		
	2122		
	14500		
	1.56		

Table.2 : Analyzed accidents and transients

Accidents	Transients	
(1) Total loss of reactor coolant flow	(5) Loss of feedwater heating	(9) Loss of load with turbine bypass valves opened
(2) Reactor coolant pump seizure	(6) Inadvertent startup of auxiliary feedwater system	(10) Loss of load without turbine bypass valves opened
(3) Control rod ejection from hot standby condition	(7) Partial loss of reactor coolant flow	(11) Control rod withdrawal from normal operation
(4) Control rod ejection from cold standby condition	(8) Loss ofoff-site power	(12) Control rod withdrawal from hot standby condition

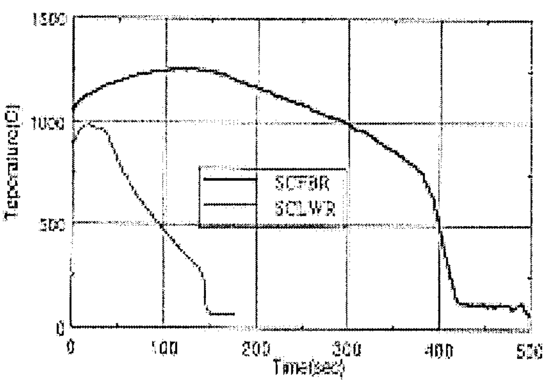
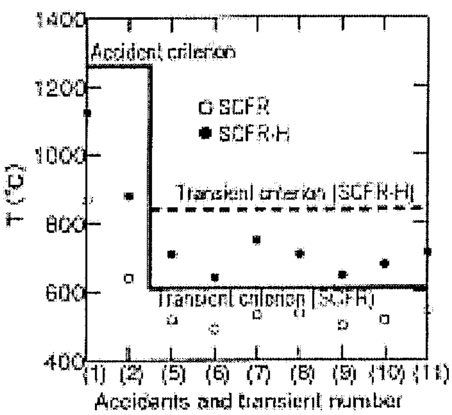


Fig.6 : Peak cladding temperature at 100% cold bread LOCA



(Numbers correspond to those in Table 2)

Fig.7 : Maximum cladding temperature of analyzed events (except for reactivity event)