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Supercritical-pressure, Once-through Cycle Light Water Cooled Reactor Concept

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The purpose of the study is to develop new reactor concepts for the innovation of light water reactors (LWR) and fast reactors. Concept of the once-through coolant cycle, supercritical-pressure light water cooled reactor was developed. Major aspects of reactor design and safety were analysed by the computer codes which were developed by ourselves. It includes core design of thermal and fast reactors, plant system, safety criteria, accident and transient analysis, LOCA, PSA, plant control, start up and stability. High enthalpy rise as supercritical boiler was achieved by evaluating the cladding temperature directly during transients. Fundamental safety principle of the reactor is monitoring coolant flow rate instead of water level of LWR. The reactor system is compact and simple because of high specific enthalpy of supercritical water and the once-through cycle. The major components are similar to those of LWR and supercritical thermal plant. Their temperature are within the experiences in spite of the high outlet coolant temperature. The reactor is compatible with tight fuel lattice fast reactor because of the high head pumps and low coolant flow rate. The power rating of the fast reactor is higher than the that of thermal reactor because of the high power density.

KEYWORDS: supercritical-pressure, light water, once-through coolant cycle, concept design, nuclear reactor, fast reactors, reactor core, nuclear power plants, safety, thermal efficiency, water cooled reactors, LWR type reactors

I. 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 to realize such innovation.[1)] The critical pressure of water is 22.1 MPa. The supercritical water does not exhibit a change of phase. Heat is effectively removed at the pseudo critical temperature, 385°C at 25 MPa 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 difference from other studies also lies in its wide scope. Almost all aspects of conceptual design have been studied by computer codes which were developed by ourselves. 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 plant system is compared with those of BWR, PWR and the supercritical fossil-fired power plant (FPP) in **Fig. 1**. The reactor concept has been developed by taking "simplicity" as guiding principle and by referring to the plant system of the once-through supercritical FPP. This paper presents the result of the concept development which took 10 years from the beginning.

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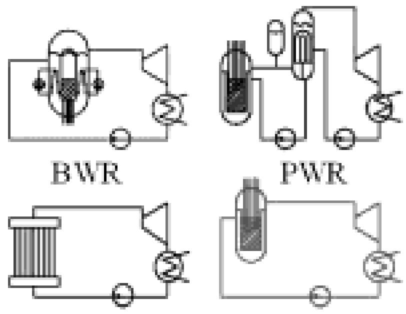

Supercritical pressure fossil-fired power plant

Supercritical pressure light water reactor

Fig. 1 Plant systems

II. 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 450°C for stainless steel (SS) and 620°C for Ni-alloy cladding
(2) Maximum liner heat rating; 39 kW/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. It is conservative from the temperature limits of those alloys in power industry. 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 was one of the critical improvements in the concept development of the once-through cycle reactor. The coolant flow rate of the once-through boiler 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 itself.[3] 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 through cycle. The operating pressure is 25 MPa. It is close to that of supercritical fossil-fired power plants, 24.2 MPa. The fuel cladding is thick enough not to be collapsed at the high pressure (27.5 MPa). The fuel rods are internally pressurized.

III. 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 2 cm. 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,

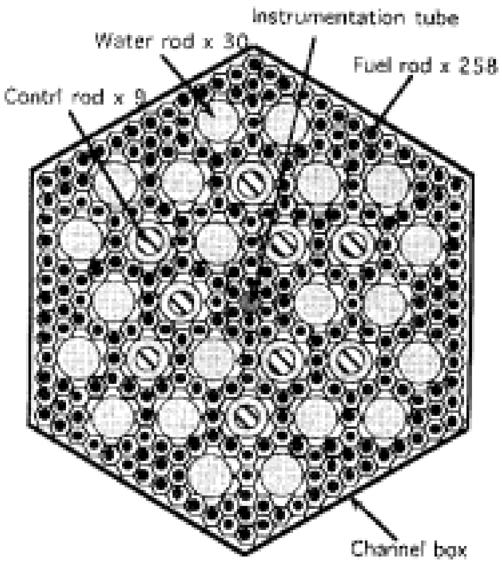

Fig. 2 Fuel assembly of SCLWR-H

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 tubes, high outlet temperature and good moderation in the upper part of the core. The demerit is the longer fuel assemblies. Fuel cladding is not collapsed in spite of the high pressure. Nickel-base alloy is taken for the design of high temperature core.

The fuel enrichment and gadolinia concentration are axially changed in three zones for flattening the axial power distribution. SRAC code system of Japan Atomic Energy Research Institute (JAERI) is used for neutronic calculation,[7] but all other computer codes were prepared for the concept development of superciritical-pressure reactors. Supercritical water is single phase fluid. This is an advantage in computational analyses, compared with the two-phase flow of LWR. 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 **Table 1**.

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 280°C. 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 1,570 MWe SCLWR-H is 33.8 cm thick and 700 t in weight, when the same steel as that of LWR is used. It is lighter than that of 1,350 MWe ABWR, 910 t. It was confirmed by the one-dimensional transport calculation that the fast neutron irradiation damage of the vessel wall is within the limit in 100 years. The numbers of inlet-outlet coolant lines are only two and small in diameter in spite of the 1,000 MWe 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 coolant system.

IV. Fast Reactor

The fast reactor, SCFR and SCFR-H adopted tight fuel lattice without water rods. The spacing between fuel rods is

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

	ABWR	SCLWR-H	SCFR-H
Thermal/electric power output (MW)	3,926/1,356	3,586/1,570	3,893/1,728
Pressure(MPa)	7.2	25.0	25.0
Thermal efficiency (%)	34.5	44.0	44.4
Cladding	Zr	Ni-alloy	Ni-alloy
Water rod/blanket flow direction	Ascending	Descending	Descending
Number of fuel assemblies	872	211	419
Effective core height/diameter(m)	3.71/5.16	4.20/3.28	3.20/3.28
Average power density (MW/m^3)	50.6	101	144[a]
Core inlet/outlet temperature (°C)	278/287	280/508	280/526
Feedwater flow rate (kg/s)	2,122	1,816	1,694
Core flow rate (kg/s)	14,500	1,816	1,694
Feedwater flow rate per electric power ($kg \cdot s^{-1} \cdot MW^{-1}$)	1.56	1.16	0.98

[a]Including blanket

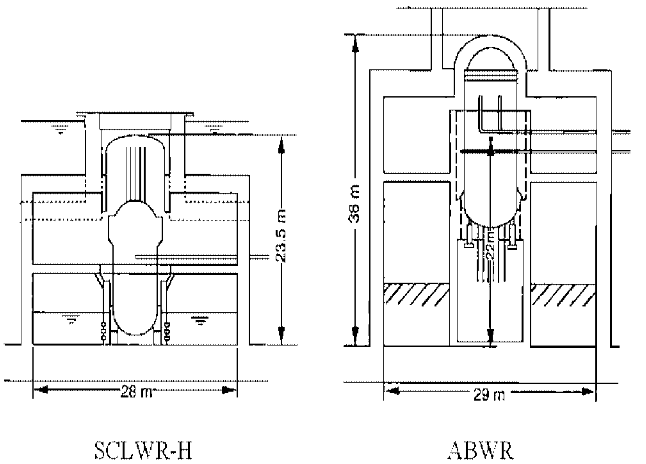

Fig. 3 Comparison of containments

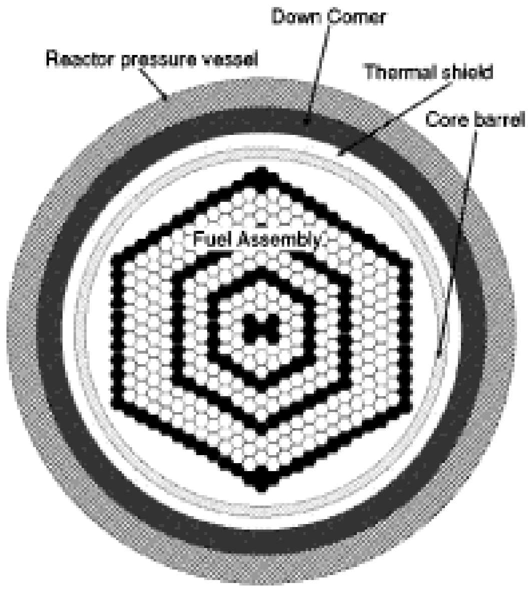

Fig. 4 Cross section of the core and pressure vessel of SCFR-H

1.3 mm. 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 and 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.[8)] 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 found in Table 1. 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 outermost layer of blankets of SCFR-H are replaced by the driver fuel assemblies, the reactor power increases to 2,017 MWe, but the conversion ratio decreases to 0.96.[9)] The breeding ratio of 1.02 was attained for the core with large fuel rods of 1.02 cm in diameter.[10)] Breeding is possible in the fast reactor, although the breeding ratio is not so high as that of the liquid metal cooled fast reactors. 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.

V. Safety

"Maintaining core flow" is the fundamental safety requirement of the supercritical water cooled reactor which has the once-through coolant system. Coolant flow rate at the inlet and the outlet of the RPV are monitored and used as the emer-

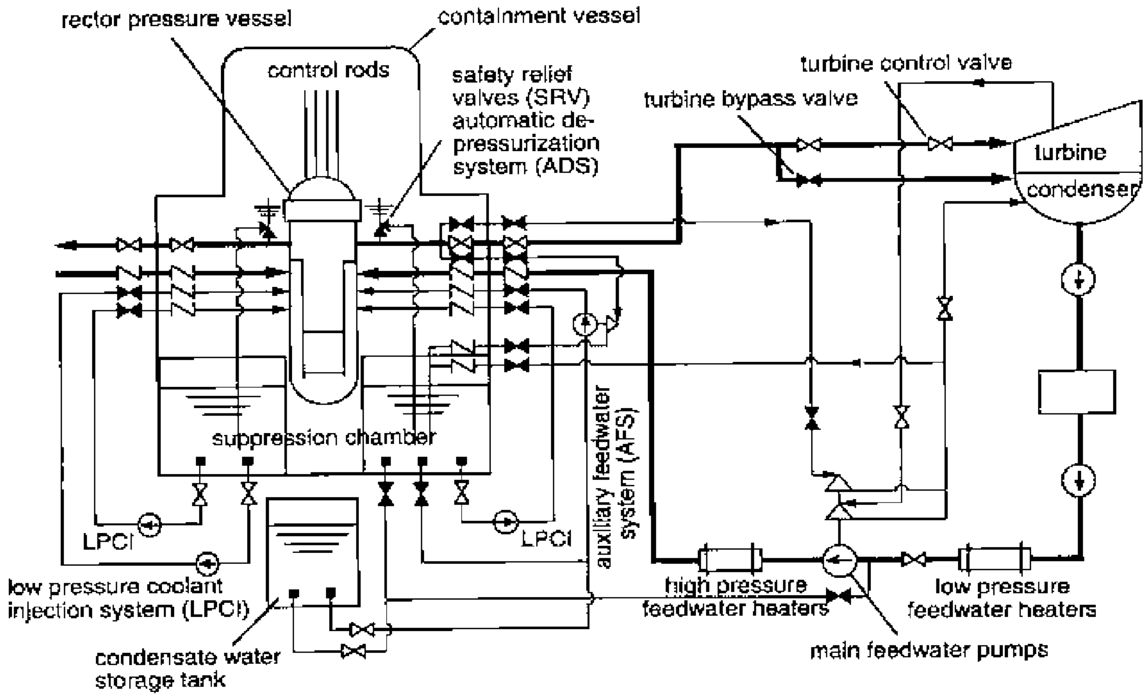

Fig. 5 SCLWR-H plant and safety system

gency signal instead of "water level" of a BWRs. 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 rated one. The system pressure is maintained at the supercritical pressure by the turbine control valves within the pressure fluctuation of 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% (1 MPa). When the system pressure becomes too low, the coolant will be released through the automatic depressurization system (ADS).

The accident criteria are

(1) Maximum cladding temperature below 1,260°C
(2) Reactor pressure below 110% of 27.5 MPa, the maximum pressure for normal operation (=30.3 MPa)
(3) Maximum fuel enthalpy below 230 cal/g (963 J/kg).

The transient criteria are

(1) Maximum cladding temperature below 610°C for the stainless steel cladding of SCFR and below 840°C for the Ni-alloy cladding of SCFR-H
(2) Maximum cladding plastic deformation below 1.0%
(3) Reactor pressure below 105% of 27.5 MPa (=28.9 MPa)
(4) Maximum fuel enthalpy below 65 cal/g (272 J/kg).

The major safety criterion of the accidents is that the maximum stainless steel (SS) cladding temperature should be below 1,260°C for avoiding core damage. It is the same as the USNRC criterion for the LWR with stainless steel cladding. Major criterion for transients is that the fuel integrity should be maintained. This is described that the maximum cladding temperature should be below 610°C for stainless steel cladding and 840°C for Ni-alloy cladding. This criterion depends on the fuel rod design and cladding strength.

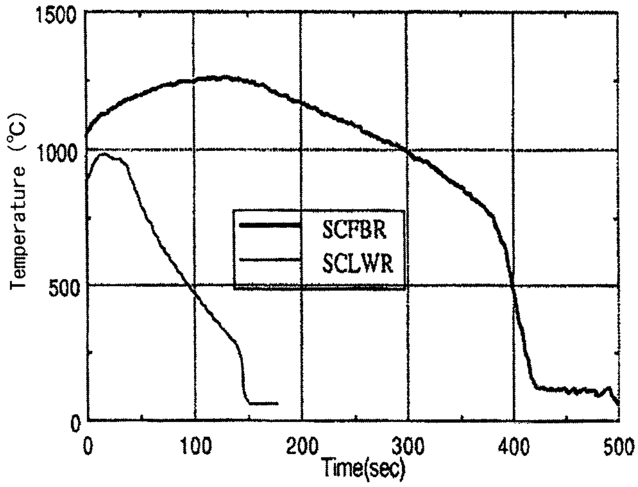

Fig. 6 Peak cladding temperatures at 100% cold leg break LOCA

The large break loss of coolant accident (LOCA) were analysed by the computer code, SCRELA.[11] The fuel cladding temperature after the 100% cold leg break is shown in **Fig. 6**. The peak temperature is below the criterion, 1,260°C. It is higher in SCFBR than in SCLWR 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 does not impose a problem of fuel integrity. Loss of coolant accident is limiting accident in the design 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

Table 2 Analyzed accidents and transients

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

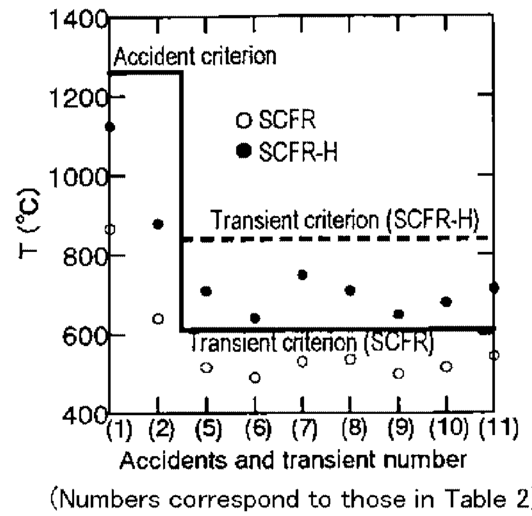

Fig. 7 Maximum cladding temperatures of analysed events except for reactivity events

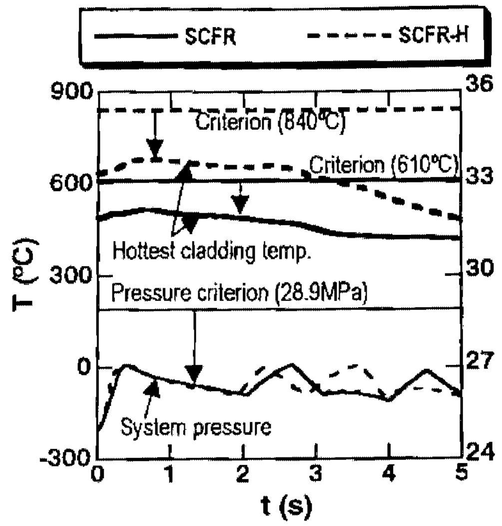

Fig. 8 Result of loss of load without opening turbine bypass valves (Transient (10))

for the analysis.[12] Nine types of accidents and transients of SCFR and SCFR-H at the supercritical pressure were analyzed (**Table 2**). The analysis showed that all cases satisfy the safety criteria (**Fig. 7**). The results of loss of load without opening turbine bypass valve is depicted in **Fig. 8**. At the beginning of the event, the system pressure rapidly increases by closing the turbine control valves until it reaches the set point of the safety relief valves (SRVs), but increases of the cladding temperature and reactor power are not so large. While the system pressure increases, the reactor power decreases due to coolant density feedback. In a once-through type coolant system, closing of the coolant outlet causes the stagnation of the core coolant flow. It increases the core coolant temperature resulting from the negative coolant density feedback. In addition, the increase of coolant density due to increase of system pressure of the SCFR and SCFR-H is smaller than that of current BWRs because no phase change occurs at supercritical-pressure. After opening the SRVs, the reactor power begins to increase because the core coolant flow is recovered. The maximum reactor power ratio reaches 105% (SCFR) of the rated power right after the beginning of the event. The loss of turbine bypass is not limiting transient for the SCFR and SCFR-H. This is an advantage of the once-through type coolant system.

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 (CDF) is assessed by simplified probabilistic safety assessment (PSA) method.[13] In order to carry out the PSA, the potential significant events which can lead to severe core damage are identified as initiating events. In the present analysis, five initiating events, large LOCA, intermediate LOCA, two categories of small break LOCAs and LOSP (loss of off-site power) are selected by considering SCFR characteristics and by acknowledging the result of NUREG 1150. Mitigation sequence for each initiating event was established with the required safety system. Event trees are constructed based on the mitigating sequences for each initiating event and referring to the PSA results of LWRs. The total CDF of SCFR is compared with the PSA result of current LWRs in **Fig. 9**. It is found that the estimated CDF is smaller than those of the U.S. BWR plants considering the characteristics of reactor system and referring to the Japanese PSA data. Accordingly, for the relative comparison between the two results, the case imposed the same initiating event frequencies as the U.S. BWR plants was calculated. The estimated CDF of the case which imposed the same initiating event frequencies shows the similar trend to the results of U.S. BWR plants. 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,

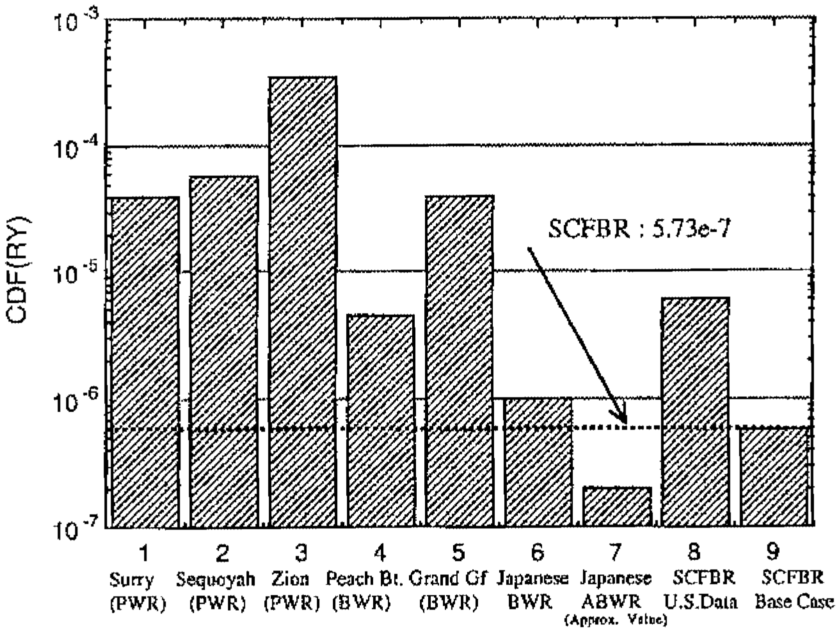

Fig. 9 Comparison of total core damage frequency with the current plants

the core damage frequency is maintained as the same level of Japanese conventional BWR because of the open coolant system of the direct cycle reactors where the coolant inventory control is not so concern as indirect cycle reactors.

VI. Plant Control, Start up and Stability

It is necessary to analyse the controllability against the perturbations of the feedwater flow rate, because all the core coolants of supercritcal-water cooled reactor, driven by the feedwater pumps, flows to the turbines without recirculating core flow. The axial coolant density change is three times larger than that of a BWR. The behaviors of SCFR are analyzed for three principal perturbation: change of the control rod position, the feedwater flow rate, and the turbine control valve opening.[14] Based on the step responses to the perturbations, the reactor control system is designed such that the pressure is controlled by the turbine control valves, the main steam temperature is controlled by the feedwater flow rate, and the core power is controlled by the control rods. To improve stability, lead compensation is needed for the main steam temperature control system. It is not appropriate to control the pressure by the feedwater flow rate like in a supercritical fossil-fired power plant because of the nuclear thermal-hydraulic coupling. Reactor behaviors with the designed control system are stable against the perturbations, although the plant is the once-through direct-cycle type, the coolant inventory is small. Reactors cooled by supercritical light water, SCFR are controllable with the control system.

The control of the thermal reactors, SCLWR and SCLWR-H was also studied. The density coefficients of reactivity of SCLWR and SCLWR-H were more than 10 times larger than that of SCFR. The reactor control systems of the thermal reactors is such that the reactor power is controlled by the main feed water pump speed and the coolant outlet temperature is controlled by the control rod. This is opposite to those of SCFR. The pressure is controlled by the turbine control valves as SCFR. High temperature supercritical steam is compressible single phase fluid, while PWR primary coolant is not. The sensitivity of coolant density with pressure is smaller than that of BWR as seen in the safety analysis. The reactor is not necessarily sensitive to the perturbations, although once-through cycle.

To estimate coolability and necessary size of SCR start-up systems, sequences and required equipment for start-up were investigated on referring to supercritical fossil-fired power plants (FPPs).[15] There are two types of supercritical fossil-fired power plant. One is the constant pressure FPP that starts up and operates at partial load at a critical pressure. The other is the sliding pressure FPP that starts up at a subcritical pressure, and operates at that pressure at partial load. The moisture content of steam to the turbine should be low enough not to damage the turbine blades at start-up. Steam can be dried in superheaters that are installed in the furnace of the FPP. SCR plant are designed to have no superheaters unlike FPPs, because installing them needs extra components and heat source and may be not economical. SCRs need to adjust steam condition without superheaters.

There are two restrictions on start-up. One is that wetness in steam should be less than 0.1% at turbine start-up, which is consistent with BWRs. The other is that the peak cladding temperature during power raising phase should be below the rated value, 450°C for stainless steel and 620°C for Ni-alloy cladding. With constant pressure start-up, the reactor starts up at a supercritical pressure. A start-up bypass system consisting of a flush tank and pressure reducing valves is needed.

With sliding pressure start-up, the reactor starts up at a subcritical pressure and the pressure increase with the load. A steam-water separator and a drain tank are needed for two-phase flow. Dryout inevitably occurs in the core at subcritical pressure. The strategy for protection of furnaces in the once-

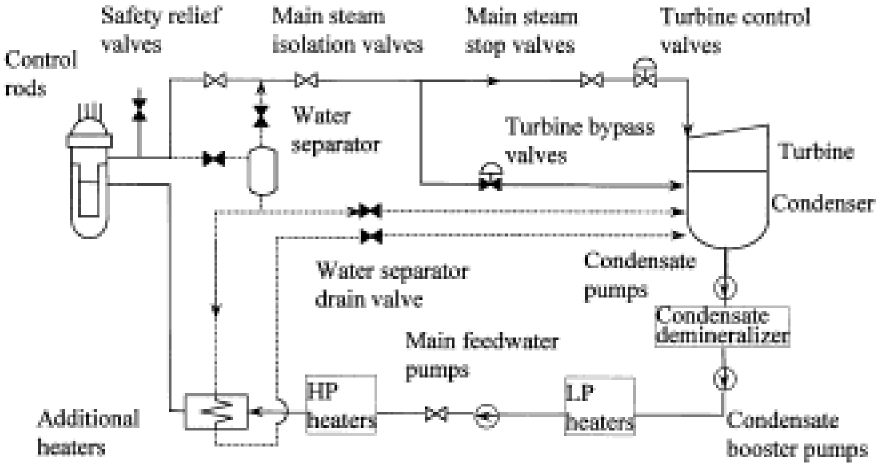

Fig. 10 Plant system for sliding pressure start-up with recirculating pump

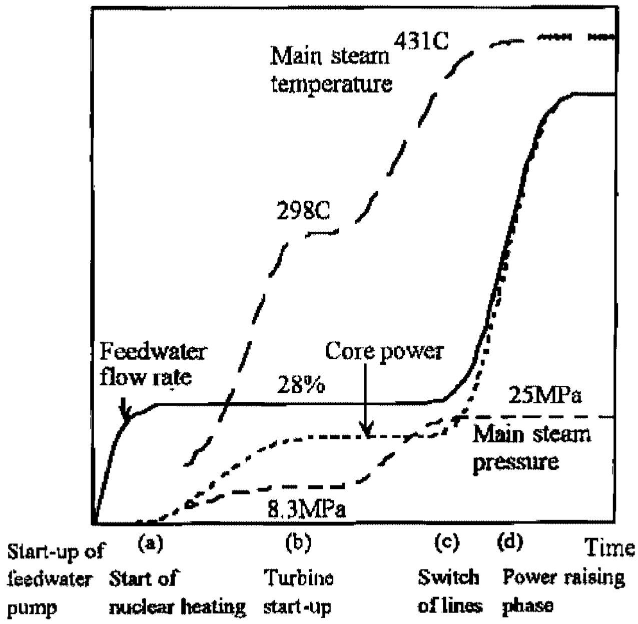

Fig. 11 Sliding pressure start-up curve

through boiler is to keep the wall temperature in the post-dryout region below an adequate value by an enough flow rate. At low pressure, burnout occurs with high quality, but the increase in cladding temperature is small. To reduce the volume of separator, it is also desirable for the core to be pressurized to a supercritical pressure with a low flow rate and a low power. The minimum flow rate will be determined from viewpoint of stability, coolability and pump performance. In the present study we assume 28%, the same as a FPP. The cladding temperature can be calculated for a certain feedwater flow rate with various core powers. The reactor is pressurized to supercritical at 28% flow rate and 20% core power. The sliding pressure start-up system is shown in **Fig. 10**. After setting the feedwater flow rate 28%, nuclear heating starts at a subcritical pressure. When the pressure of the core reaches an adequate value, saturated steam from the separator flows to the turbines. After start-up of the turbines, the core is pressurized to a supercritical pressure with a core power below 20%. The reactor power increases with the feedwater flow rate as the constant pressure start-up scheme. The start up curve is schematically shown in **Fig. 11**. The sizes of the components required for the start-up systems were assessed based the present Japanese Industry Standard for vessel materials. To simplify the plant system and to reduce the component size, sliding pressure start-up with steam separator in bypass line is the best.

A thermal-hydraulic stability and a thermal-nuclear coupled stability are analyzed by developing a computer code at supercritical pressure.[16] Using this code, stability of full and partial-power reactor operating at supercritical pressure are investigated by the frequency-domain analysis. The same stability criteria as LWR are applied. The thermal-hydraulic stability of SCLWR-H and SCFR-H satisfies the criteria with a reasonable orifice loss coefficient. The decay ratio of the thermal-nuclear coupled stability in SCFR-H is almost zero because of a small coolant density coefficient of the fast reactor. The decay ratio of the thermal-nuclear coupled stability is 0.028 at 100% power in SCLWR-H. It is found that the thernal-hydraulic stability is sensitive to the mass flow rate strongly and the thermal-nuclear coupled stability to the coolant density coefficient. The bottom power peak distribution makes the thermal-hydraulic stability worse and the thermal-nuclear stability better.

VII. Discussion

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.[17] Simplicity of plant system in the once-through cycle is the other advantage.[18]

The supercritical pressure once-through cycle is more compatible with tight-fuel-lattice fast reactor core than LWR because of the high head pumps and small coolant flow rate, approximately one eighth of LWR. The pumping power is small. The advantage of high power density of fast reactor over thermal reactors will increase the economic competitiveness of the fast reactor over LWR.

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

There exist abundant experiences of materials in supercritical thermal boilers, but not under radiation exposure. The fundamentals of supercritical water chemistry in reactor environment should be understood. Testing of cladding materials in autoclaves and then in-pile loops are necessary. Corrosion product (CP) transport to turbines should also be studied in relation with water chemistry. It is necessary to take into account of the characteristics of the supercritical water and once-through cycle system where the coolant flows through the core only several seconds and all corrosion product can be collected after condensation. Thermal hydraulic testing of a fuel assembly is necessary for assessing the accuracy of the heat transfer calculations. In the present study, the heat transfer coefficients which are derived for a smooth tube is used for the design. Heat transfer will be improved in a fuel assembly. Hydraulics of core internals with water rods is also necessary to be assessed.

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VIII. Conclusion

The concept of once-through cycle supercritical pressure light water cooled reactors was developed by computer analysis. Fundamental guidelines in designing the reactor are given. The critical heat flux criterion as that of LWR is eliminated from the design criteria by developing the method and data base which predict the cladding temperature during transients. This makes it possible to design high temperature reactor which has large enthalpy rise in the core. Fundamental safety principle of the reactor is monitoring “coolant flow rate” instead of “water level” of LWR. The safety design and criteria are similar to those of LWR. The behavior at loss of load transient without opening turbine bypass valve is not so severe in contrast with that of BWR. Probabilistic safety assessment showed that the core damage frequency is similar to that BWR, although no natural circulation is established at total loss of feedwater flow in the once-through cycle reactor. This is because of the open coolant system of the direct-cycle reactors where the coolant inventory control is not so concern as the indirect-cycle reactors. The sliding pressure start up is better than the constant pressure startup from the weight of the start-up system. The steam-water separators should be set up in the bypass line. The reactor system is compact and simple because of the high specific enthalpy and the elimination of recirculation and steam-water separation systems. The advantages in manufacturing and reliability are originated from the experiences of major components in supercritical fossil-fired power plants and LWR. The once-through cycle is compatible with the tight-fuel lattice fast reactor core because of the high head pumps and the small coolant flow rate.

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