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SAFETY ANALYSIS OF A SUPERCRITICAL PRESSURE, LIGHT WATER COOLED AND MODERATED REACTOR WITH DOUBLE TUBE WATER RODS

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Abstract—The supercritical pressure, light water cooled and moderated reactor (SCLWR) has once-through cooling system. All feedwater which cools the reactor core flows to the turbines. This paper summarizing the safety analysis of the SCLWR with double tube water rods. The plant system is simple but no natural circulation is established at the loss of feedwater flow. The coolant inventory in the reactor pressure vessel is small. The coolant density coefficient is approximately twice as large as that of the BWR. A computer code (SPRAT) was developed to analyze SCLWR behavior against major accidents and transients at supercritical pressure. In loss of flow events such as loss of off-site power, the flow coast down time should be larger than 10 sec. for avoiding the deterioration in heat transfer. In the flow-excess event such as inadvertent start of the auxiliary feedwater pumps, the power increases approximately 25% by coolant density feedback. In the overpressurization transient such as generator load rejection, the power does not increase even if scram fails. This is because flow stagnantion raises coolant temperature and coolant density change at overpressurization is small in supercritical pressure. The reactivity-induced event such as control rod ejection, is not severe because of the small reactivity ingress. In the loss of coolant accident, the double tube water rods delay the reflood of the core. The core is heated up rapidly because of the small heat capacity and tight lattice pitch of the fuel rods. All analyzed accidents and transients satisfied the criteria, and the feasibility of the reactor was confirmed from the safety point of view.

1. INTRODUCTION

The advantages of the supercritical pressure, light water cooled reactors are the high thermal efficiency and simple plant system (Oka Y., Koshizuka S. and Yamasaki T., 1992) (Oka Y. and Koshizuka S., 1993) (Okano Y., Koshizuka S. and Oka Y., 1994a 1994b 1996a 1996b 1996c). The supercritical pressure light water cooled and moderated reactor (SCLWR) was designed as the thermal spectrum reactor by using the double tube water rods (Okano Y., Koshizuka S. and Oka Y., 1996b 1996c).

Water does not exhibit a change of phase above the critical pressure of 22.1 MPa. Specific heat, density, enthalpy, kinetic viscosity coefficient of supercritical water at 25.0 MPa are shown in Fig.1. Heat capacity reach a maximum at the pseudocritical temperature (386 °C). Density, enthalpy and kinetic viscosity coefficient change

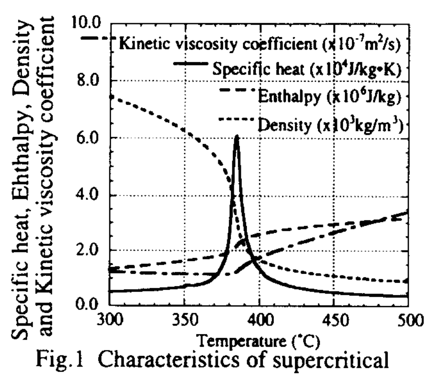

Fig.1 Characteristics of supercritical pressure water

largely but continuously. It is different from the boiling phenomenon at subcritical pressure, where heat capacity, density and enthalpy change discontinuously at the boiling temperature. Supercritical fluid can be treated as single phase flow.

The heat transfer of supercritical water deteriorates under high heat flux and low coolant flow rate condition. The temperatures of the coolant and at the tube surface are shown in Fig.2 as a function of mass flow rate (Ackerman J. W., 1970). Deterioration in heat transfer occurs only in the case of mass flow rate of 540 kg/m²s. The tube surface temperature rises where the coolant temperature is a little lower than the pseudocritical temperature, 405 °C at 31.0 MPa. The heat transfer deterioration is much milder phenomenon than the dryout of the subcritical water cooling because the cladding surface temperature does not increase rapidly and the heat transfer coefficient recovers in the downstream. In this study, Yamagata's correlation was used for evaluating the deterioration heat flux (Nishikawa K. *et al.*, 1971 1972) (Yamagata K. *et al.*, 1972).

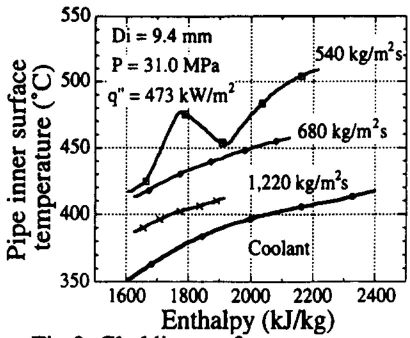

Fig.2 Cladding surface temperature of supercritical water cooling

$$q''_{limit} = 2.035 \times 10^2\, G^{1.2} \quad (1)$$

where

q''_{limit} : Deterioration heat flux (W/m²)

G : Mass velocity (kg/m²s)

This correlation is given as a function of the 1.2 power of mass flow rate, $G^{1.2}$. On the contrary, the criteria for the heat flux at the subcritical water cooling has a linear relationship with approximately 0.3-0.5 power of mass flow rate, $G^{0.3\text{-}0.5}$ (J. S. M. E. Data Book, 1986). Hence, the heat transfer criterion is more sensitive to the change of coolant flow at the supercritical water cooling than the subcritical one.

The SCLWR has the once-through cooling system, where all coolant are driven by feedwater pumps, flowing through the reactor core, and fed to turbines directly. By using supercritical water cooling, recirculation system, steam separator and dryer of the current BWR can be eliminated. The coolant system has only two lines for 1,100MWe class power plant. Thus, the SCLWR system is much compact and simple comparing with the current LWRs.

The characteristics of the SCLWR are summarized in Table 1. The pressure is 25.0 MPa. The effective core height is 3.7 m. and equilibrium diameter is 3.06 m. Fuel rod diameter is small, 0.80 cm, and the triangular lattice of fuel rods is tight, 0.95cm. The fuel is UO_2 and its average enrichment is 5.12 wt%. The stainless steel was selected as the cladding material to withstand the high cladding surface temperature of 450 °C. The inlet and outlet coolant temperature is 300 °C and 408 °C respectively. The coolant density changes large axially, 743 kg/m³ at the inlet and 148 kg/m³ at the outlet. The double tube water rods, which contain the inner and outer tubes, are used for enhancing the moderation. In the double tube water rods, whole coolant flow into the inner tube from the lower plenum, turn downward to the outer tube at the top and is discharged into the coolant channel between fuel rods at the bottom. The hydrogen to heavy metal

Table.1 SCLWR characteristics

Pressure (MPa)	25.0
Effective height / Equivalent diameter (m)	3.7 / 3.06
Fuel rod diameter / Lattice pitch (cm)	0.80 / 0.95
Pellet / Cladding material	UO_2 / Type316 S.S.
Fuel enrichment (average) (wt%)	5.12
Hydrogen / Heavy metal ratio (average)	3.4
Discharge burnup (GWd/t)	42.3
Coolant inlet / outlet temperature (°C)	300.0 / 408.2
Coolant inlet / outlet density (kg/m³)	743.3 / 148.2
Maximum fuel centerline (°C)	1,809
Maximum cladding surface temperature (°C)	450
Coolant density coefficient (pcm/(kg/m³))	40
Doppler coefficient (pcm/K)	-2.1
Number of Fuel assemblies	187
Average power density	105
Feed water flow rate (kg/s)	2,126
Core power (Thermal / Gross electric)	2,856 / 1,180
Thermal efficiency	41.3%

ratio (H/HM) is approximately 3.4 in average, where the the core is a little under moderated in comparison with the optimum moderated condition. The average discharge burnup is 42.3 GWd/t. The coolant density coefficient of the operating condition is 40 (pcm/(kg/m^3)), which is approximately twice as much as that of the current BWR and ten times of the supercritical pressure light water cooled fast breeder reactor (SCFBR). The Doppler coefficient is approximately -2.1 (pcm/K) at the operating condition, which is nearly the same as those of the current LWRs. The minimum deterioration heat flux ratio (MDHFR), which is defined as the ratio of the deterioration heat flux to the fuel surface heat flux at the deterioration point, was used as the criterion for the core design. The coolant flow at the low power region is limited to enhance the outlet coolant temperature and the MDHFR is 1.30. The thermal efficiency is 41.3 %. The electric power is 1,180 MWe.

2. SAFETY CHARACTERISTICS

The diagram of plant systems of the BWR, PWR and SCLWR are compared in Fig.3. Natural circulation through the core can be established in both BWR and PWR, when the pumps are stopped. The SCLWR is the once-through type plant. No natural circulation is established when the main feedwater flow is stopped. The flow abnormality directly affects the heat balance in the core. SCLWR looks very sensitive to the flow induced perturbation. The coolant density coefficient is approximately twice as much as that of the BWR and ten times of that of the SCFBR. The axial coolant density change in the core is large like the BWR. The coolant density will increase at overpressurization.

The rod control clusters (RCCs) are adopted as used in the PWR. The RCC driving mechanisms are mounted on the top of the RPV and RCCs are inserted from the top of the core. The control rod are ejected quickly when breaking the support mechanism of the RCC.

The reactor pressure vessel (RPV) of the SCLWR is similar to that of the PWR as shown in Fig.4. The coolant inventory in the RPV is smaller than that of BWR. The fuel rod diameter is small, 0.80 cm, and the heat capacity is smaller than that of the SCFBR. The fuel lattice is triangular and its pitch is tight, 0.95cm. The thermal hydraulic diameter of coolant channel between fuel rods is 0.69cm, which is also smaller than that of the SCFBR of 0.93 cm (Lee J. H., Koshizuka S. and Oka Y., 1996). Heat removal by reflooding the SCLWR core in the loss of coolant accident (LOCA) is more difficult than LWR. In addition, the double tube water rods should be filled with coolant before the fuel rods. The effect of the double tube water rods on the LOCA behavior had not been considered in the past LOCA analysis of the supercritical-pressure, light water cooled reactors (Lee J. H., Koshizuka S. and Oka Y., 1996).

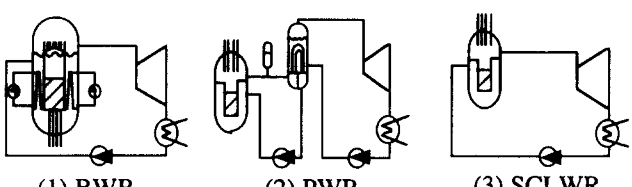

Fig.3 Plant system diagram of BWR, PWR and SCLWR

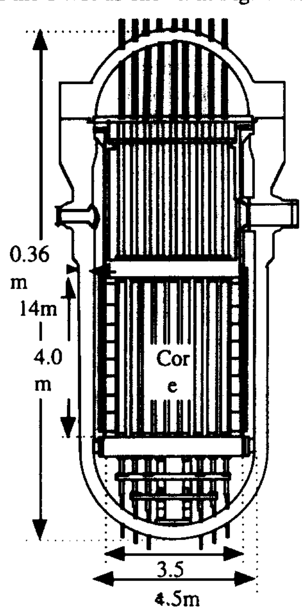

Fig.4 Reactor Pressure Vessel

3. SAFETY SYSTEM

The basic safety requirement for the SCLWR is to maintain the coolant flow in the core. It is necessary to maintain feedwater flow from the cold leg and to keep the hot legs open. The safety systems are designed as shown in Fig.5 (Okano *et al.* 1995a 1995b) (Okano Y., Koshizuka S. and Oka Y., 1996a). Mass flow rates at the cold and hot legs are used for the emergency signal instead of the "water level" of a BWR. Depending on the level of abnormality in the mass flow rates, the following actions are taken :

Flow level 1 (90% of nominal flow rate) : reactor scram,

Flow level 2 (20% of nominal flow rate) : actuation of auxiliary feedwater system,

Flow level 3 (10% of nominal flow rate) :actuation of automatic depressurization system and low pressure coolant injection system.

The signal for actuating the reactor scram is released to prevent the core from power-flow mismatch when the mass flow rate at the inlet decreases below 90% of the nominal flow. It is also released when the reactor power increases above 110% of the nominal power. In the safety analysis, the control rods are assumed to take 3.0 sec for full insertion after the actuation. The delay time for signal detection and processing is also considered, which differs on the detection method ; 0.5 sec. for neutron flux, 1.0 sec. for flow rate, 2.0 sec for pressure. The scram reactivity is changed depending on the burnup and core conditions. The maximum value is used for the analysis.

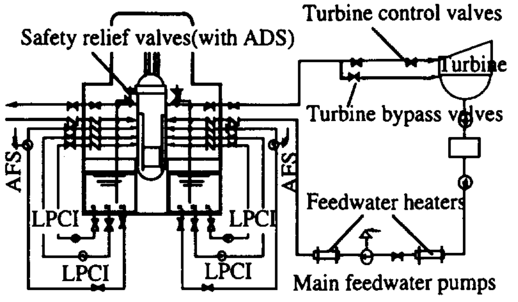

Fig.5 SCLWR safety systems

Two turbine-driven high-pressure auxiliary feedwater systems (AFWSs) are directly connected to the cold legs for keeping the core flow rate when the main feedwater system is tripped. The pump head is assumed 26.0 MPa. The capacity of each system is decided from the safety analysis enough to remove the decay heat. They also serve as the reactor core isolation cooling system. The coolant temperature driven by the AFWS is assumed 30 °C.

Four low pressure coolant injections (LPCIs) are equipped for making up the coolant inventory after the blowdown. Two of them are connected to the cold legs and the others directly to the downcomer of the RPV. In the LOCA, two out of four of the LPCIs will be actuated.

The turbine control valves are used to regulate the pressure at the steady state condition. The following pressure control systems are used according to the levels of abnormality in the pressure.

Pressure high level 1 (above 26.0 MPa) : turbine bypass valves,

Pressure high level 2 (above 26.5 MPa) : safety relief valves.

When the turbine control valves cannot regulate the pressure and it increases above level 1 of high pressure, 26.0 MPa, the turbine bypass valves are actuated. The safety relief valves (SRVs) start releasing the hot coolant at the level 2 of high pressure, 26.5 MPa. There are five SRVs on the main steam pipes. Each SRV has the different preset point of pressure for opening and closing. In the flow, pressure and reactivity-induced accident and transient analysis, all SRVs are assumed to have the same capacity in the mass flow rate, 540 (kg/s), which is calculated from the critical flow at the operating pressure of 25.0 MPa.

The following safety systems are actuated according to the decrease in the pressure ;

Pressure low level 1 (below 24.0 MPa) : reactor scram,

Pressure low level 2 (below 23.0 MPa) : automatic depressurization systems and low pressure coolant injections.

When the pressure cannot be maintained by the turbine control valves and decreases below the level 1 of low pressure, 24.0 MPa, the signal for the reactor scram is released. Heat transfer deteriorates remarkably around the critical pressure of 22.1 MPa. Hence, the automatic depressurization systems (ADSs) are actuated and decreases the pressure to subcritical at the level 1 of low pressure. And LPCIs are assumed to start removing the decay heat for core cooling after decreasing the pressure below 0.85 MPa.

4. ANALYTICAL MODEL AND PROCEDURE

A computer code for analyzing the accidents and the transients at supercritical pressure (Supercritical Pressure Reactors Accidents and Transients analysis code, SPRAT) was prepared. The SPRAT was prepared for the safety analysis of the SCLWR using water rods based on the previous computer code which had been developed for the flow and pressure-induced safety analysis of the supercritical-water-cooled reactors (Okano Y. *et al.*, 1995 1996) (Okano Y., Koshizuka S. and Oka Y., 1996a). The previous computer code can only deal with the flow and pressure-induced accidents and transients at supercritical pressure. In the SPRAT, the model for thermal calculation for the fuel rod is improved for treating the reactivity-induced events accurately. The SPRAT can deal with flow, pressure and reactivity induced transients and accidents at supercritical pressure.

The core was modeled as a single channel as shown in Fig. 6. The core is axially divided into ten nodes. The volume of the main steam pipes is included in that of the upper plenum. The volume of the cold leg and downcomer is included in that of the lower plenum. Each plenum was modeled by a single node. The behavior of the hottest single channel was analyzed. The temperatures of the fuel rod and the water rod tubes are calculated by the equation of non-steady heat conduction. The power distribution in the fuel pellet can be changed depending on the burnup and the operating condition, such as hot full power, hot standby and cold zero power conditions. The coolant channel among the fuel rods is modeled as the single channel (Okano Y., Koshizuka S. and Oka Y., 1994a). The heat transfer between fuel rods and coolant channel is calculated. The double tube water rod is also modeled as the single channel. The heat transfer between coolant channels and water rod and between the outer and inner tubes of water rod are calculated. Mass and energy conservation equitations for the coolant are solved. This calculation proceeds from the inlet to the outlet following the flow direction ; inlet pipes, downcomer and lower plenum, inner and outer tubes of water rods, coolant channel among fuel rods, upper plenum and outlet pipes.

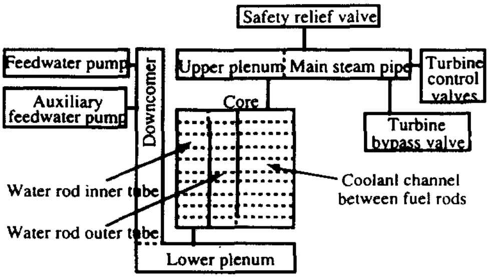

Fig.6 Calculation model for safety analysis

The computer code consists of the thermal-hydraulic and nuclear calculation. The thermal hydraulic calculation is carried out based on the following assumptions ;

1) The pressure is uniform in the RPV.

2) The axial power distribution follows the cosine shape.

3) The inlet coolant flow rate is determined from those of the main feedwater pumps and the AFWSs and it is given as the inlet boundary condition.

4) The outlet coolant flow rate is determined from opening of the turbine control, bypass and safety relieve valves, and it is given as the outlet boundary condition.

5) The heat transfer is calculated using Dittus-Boelter formula, which gives lower heat transfer coefficient than the experimental values at supercritical water cooling.

The nuclear calculation is carried out as follows :

1) The point kinetics equation with six delayed neutron groups is used.

2) Decay heat is calculated using two groups approximation of the ANS+20% evaluation.

3) The axial neutron distribution is the same as the axial power profile.

4) The reactivity feedbacks of the coolant density and the Doppler are considered.

5) The reactivity is weighted by the axial distribution of square of neutron flux.

In the SPRAT, the calculation proceed as follows. Firstly, initial conditions (mass flow rate, coolant temperature and density, cladding and fuel pellet temperature, pressure and maximum liner heat rate of the fuel) are given. The mass flow rate and the coolant temperature at the inlet are given as a function of time. Thermal hydraulic calculation is carried out in the flow direction using the mass and energy conservation equations. The temperature distribution of the fuel is calculated from the generated heat and heat transfer. The calculated mass flow rate at the outlet is compared with the mass flow rate determined from the opening of the turbine control valves. If these is any difference between these two mass flow rates, the pressure is changed and the calculation returns to the previous step. Finally, reactivity feedback is calculated using the average coolant density and fuel temperature. The power is calculated from the point kinetic equation and the Decay heat.

5. SAFETY CRITERIA

The criteria used for the safety analysis are ;

Accidents : The cladding surface temperature of stainless steel stays below 1260 °C,

The system pressure should be below 33 MPa.

The fuel enthalpy should be below 230 cal/g_{UO2}.

Transients : The MDHFR should be above 1.0.

The system pressure should be below 30 MPa.

The fuel enthalpy should be below 170 cal/g_{UO2}.

The safety criterion for the accident is to avoid large core damage. The temperature of the Zircaloy cladding at LOCA should be below 1,200°C for satisfying the criterion. The temperature criterion at the LOCA for the stainless steel clad PWR fuel in USA was assessed by USNRC in comparison with that of the Zircaloy, which was reported in NUREG report (Franklin D. C. Jr., 1976). The stainless steel is both mechanically and chemically more stable up to 1,200 °C. The report concluded that a peak clad temperature criterion of 1,260 °C was shown to be appropriate when evaluating stainless steel clad fuel. The maximum system pressure of 30 MPa is 1.2 times larger than that of the normal operating condition of 25 MPa, which is decided with considering the pressure criteria of the LWRs. The maximum fuel enthalpy criterion of 230 cal/g_{UO2} is the same as that of LWRs.

The deterioration in heat transfer occurs at high heat flux and low flow condition at supercritical-water-cooling. Yamagata's correlation was assumed to be applied at the coolant enthalpy of 1.675×10^6 (J/kg) which corresponds to the pseudocritical temperature at 25.0 MPa. This correlation was obtained by the experiment using the single smooth tube. The critical heat flux in the fuel bundle will be higher due to the turbulence. Furthermore, it should be noted that the cladding surface temperature does not rise so much if the heat transfer deterioration occurs for a short time. The basic safety requirement for transient is to avoid the core damage and that the reactor can be returned to operation again. The criterion for the transient, MDHFR above 1.0 was decided considering these characteristics of the deterioration in heat transfer of the supercritical water cooling. The criterion for the system pressure of 30 MPa is 1.1 times larger than the operating pressure. The fuel enthalpy are the same as that of LWRs.

6. CALCULATION RESULT

The analyzed safety events were grouped into accidents and transients (Okano Y. *et al.*, 1995 1996). The following typical safety events are analyzed in this study.

I) Loss of off-site power (transient),
II) Inadvertent start of the AFWSs (transient),
III) Loss of feedwater heater (transient),
IV) Generator load rejection (transient),
V) Control rod ejection (accident),
VI) Loss of coolant (accident).

5-1) Loss of Off-Site Power (transient)

The off-site power of the SCLWR is postulated to be lost. There is no natural circulation in the once-through. All electrically powered pumps are tripped and the feedwater flow decreases and finally the main feedwater pumps which are driven by the turbines are tripped.

The calculation result is depicted in Fig. 7. When the off-site power is lost, the actuation signals for the scram and AFWSs are released. The scram starts after the delay time for signal processing of 0.5 sec. The control rods are completely inserted at 3.5 sec. The AFWSs start at 2 sec. and feed 16% of the nominal flow rate after 3 sec. The driving power is postulated to be lost at 10 sec. and all feedwater pumps start coast down. The AFWSs are equipped with flywheels. The feedwater linearly decrease and lost at 20 sec.

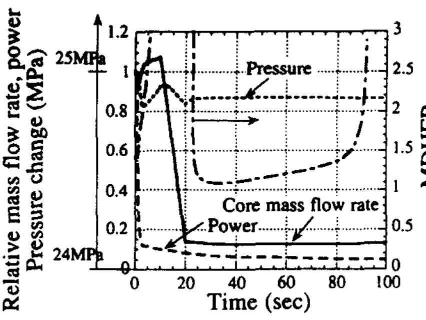

Fig.7 Loss of off-site power

The power decreases by the scram and the MDHFR increases gradually until approximately 10 sec. The MDHFR decreases quickly after stopping the main feedwater pumps and continuously decreases until the coolant flow rate in the core recovers enough to remove the decay heat. Its minimums value are 1.09 at 38 sec. and satisfy the safety criteria. The coast down time of the main feedwater pumps of 10 sec. and AFWSs capacity of 16% were determined to satisfy the criteria.

5-2) Inadvertent start of the auxiliary feedwater pumps (transient)

Two AFWSs are postulated to start inadvertently. The calculation result is depicted in Fig. 8. The inlet mass flow rate increases 116% in 1.0 sec. The power increases continuously because of the feedback of the coolant density. The power reaches 110% at 0.8 sec. and the signal for the scram is released. The power starts to decrease at 1.2 sec. because of the increase in coolant temperature. The control rods are begun to be inserted at 1.3 sec. and completed at 4.3 sec. The MDHFR stays approximately 1.56 until 2 sec. The power decrease by the scram and the pressure also decreases until 5 sec. The MDHFR increases after the decrease in the power by the scram.

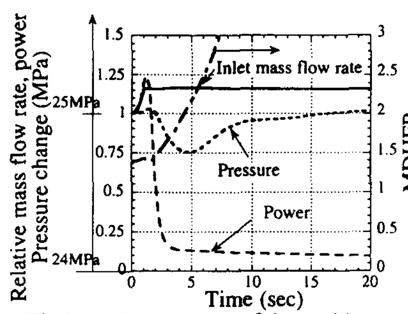

Fig.8 Inadvertent start of the auxiliary feedwater pumps

The increase in the coolant flow in the core affects coolant density quickly in the once-through type plant. The power increases quicker and larger than that of the SCFBR because of the large coolant density coefficient. Increase in the core flow becomes one of the reactivity-induced events in the SCLWR. In the SCLWR, the maximum reactivity insertion by the coolant density feedback is approximately $0.23. The criterion for the pellet enthalpy should be satisfied in the reactivity-induced events. The maximum fuel enthalpy is 100 cal/g, which is only 3 cal/g higher than the initial value, and satisfies the criterion of 170 cal/g.

5-3) Loss of feedwater heating (transient)

One stage of the feedwater heater is lost. It is considered to decrease in the inlet coolant temperature by 35 °C. Cold coolant at the inlet flows into the lower plenum. It increases coolant density and decreases mass flow rate in the lower plenum. The mass flow rate in the core also decreases and the fuel and coolant temperatures increase. With increasing coolant temperature, coolant density and reactivity decreases because of the negative coolant density feedback. Hence, the power stayed around 93% of the nominal one until 7 sec. as shown in Fig. 9. The cold coolant flows into the core after 8 sec. The power gradually increases and reaches 110% at 10.3 sec., when the scram signal is released. The scram starts at 10.8 sec and completes at 13.8 sec. The MDHFR decreases to 1.17 at 10.8 sec, but they are higher than the criterion.

In the BWR, the feedwater is mixed with the recirculation coolant. Hence, the coolant temperature in the core decreases but not so much at the loss of feedwater heating. In the SCLWR, such mixing does not occur because there is no recirculation coolant. Hence, this transient is one of the flow-induced events in the once-through type plant. This transient was thought to be severe in the SCLWR, but it was not.

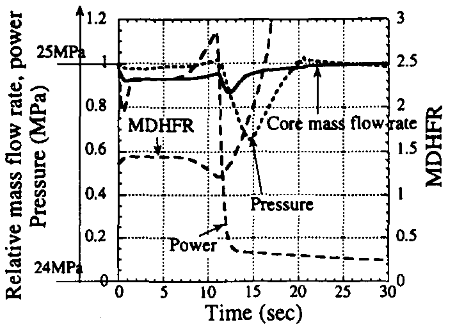

Fig.9 Loss of feedwater heating

5-4) Generator load rejection (transient)

The generator load is postulated to be rejected suddenly. The turbine control valves are closed to prevent the turbine from damaging in 0.07 sec. The coolant flow decreases and the pressure increases quickly. The signal for opening the turbine bypass valves is released at 0 sec. but in failure. The calculation result is shown in Fig. 10. The core flow rate decreases and the pressure increases continuously. The scram signal of level 2 of high pressure is released at 0.16 sec. The maximum pressure is 27.3 MPa at 0.38 sec. The SRVs are open according to the preset pressure for opening. When four SRVs are open at 0.4 sec., the pressure starts to decrease. The pressure oscillates after 1.8 sec. with opening and closing the SRVs. The core flow rate decreases and the coolant temperature increases with closing the turbine control valves. Hence, the coolant density does not change so much in spite of the quick increase in the pressure. The power changes little until the start of control rod insertion at 2.2 sec. The MDHFR decrease to 1.13 at 0.2 sec., and satisfy the criteria.

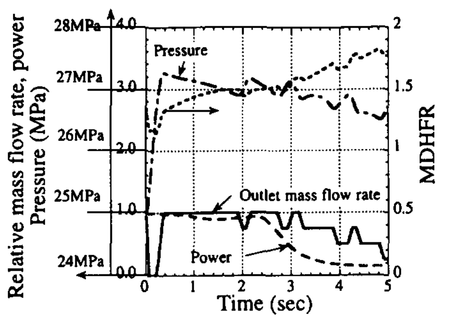

Fig.10 Generator load rejection

In the BWR, the pressure-induced transient is severe. In the SCLWR, the decrease in the core flow rate compensates the pressure rising in the pressure in the generator load rejection. This transient is not severe because of the two times larger coolant density coefficient than the BWR.

5-5) Generator load rejection (Anticipated transient without scram)

It is assumed that the scram fails at the generator load rejection transient. The SCLWR behavior is nearly the same as that with scram within 2.2 sec. as shown in Fig.11. After 2.2 sec., the pressure oscillates between 26.9 MPa and 27.1 MPa with opening and closing one of the SRVs. The power does not change so much and stays around 100% of steady states condition. The maximum pressure is 27.3 MPa. The MDHFR is above 1.13, and satisfy the criteria. Anticipated transient without scram (ATWS) does not cause severe power rise in the SCLWR.

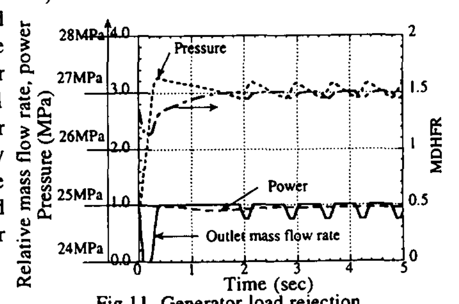

Fig.11 Generator load rejection
(Anticipated transient without scram)

5-6) Control rod ejection (accident)

The control rod ejection accident is important for the SCLWR, since the control rod worth is higher than that of a PWR for compensating the burnup reactivity without the chemical shim. The control rod is inserted from the top of the core and its ejection causes the reactivity-induced accident. The control rod ejection was estimated to take a very short time of 0.10 sec. like that of the PWR. One control rod cluster is postulated to be ejected. The highest reactivity ingress is assumed conservatively \$0.76, which corresponds to the reactivity of the one control rod stuck. The core power increases quickly until the reactivity of Doppler and coolant density feedbacks become effective. The change of the pellet enthalpy is shown in Fig.12. The maximum fuel enthalpy is 102 cal/guo2, where the initial fuel enthalpy is 97 cal/guo2. The maximum cladding surface temperature is only 470 °C. Control rod ejection at the HZP and CZP were also analyzed, and the criteria were satisfied. When the Doppler coefficient is twice and half of the normal one, the maximum fuel enthalpy is 103 and 101 cal/guo2, respectively. The enthalpy change at postulated reactivity insertions of $1, $2, $2.5 and $3 are also shown in Fig. 9. The inserted reactivity in the control rod ejection accident is allowable up to approximately $2.5.

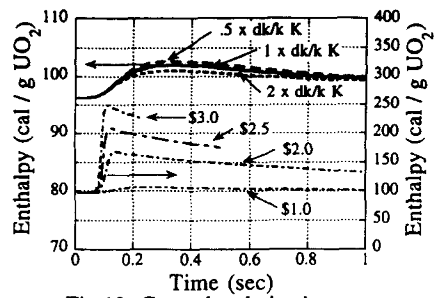

Fig.12 Control rod ejection

5-7) Loss of coolant accident

The loss of coolant accident of the SCLWR with double tube water rods was analyzed. It was concerned that the reflood during the LOCA is delayed due to the double tube water rods and the tight lattice. In the SCLWR, the cold leg break LOCA is more severe than the hot leg break (Lee J. H., Koshizuka S. and Oka Y., 1996). After the break of the cold leg, the coolant flow direction is reversed. The heat transfer coefficient decreases and fuel temperature rises quickly at the flow stagnation. The coolant does not flow into the core until the level in the water rod and downcomer reaches the core height of 3.7 m.

The computer code, SCRELA, had been developed for the LOCA analysis of supercritical water cooled reactors (Lee J. H., Koshizuka S. and Oka Y., 1996). In the present study, the model of the water rods was added to the SCRELA. The water rods were dealt with as a single node between the lower plenum and the active core. The heat transfer between the outer tube of water rods and surrounding coolant channel are taken into account. The following estimations were used to consider the above effects. (1) The core reflood does not start until the double tube water rod is quenched and the water level reaches the core height of 3.7 m and coolant fills the inner tubes completely, (2) The radiation heat transfer to the structural material, such as the reactor pressure vessel, core barrel and core support plats, was neglected. As the LPCI, there are two motor-driven pumps and two turbine driven pumps. The capacity of each pumps is 805 (kg/s). Only two of them were postulated to be actuated.

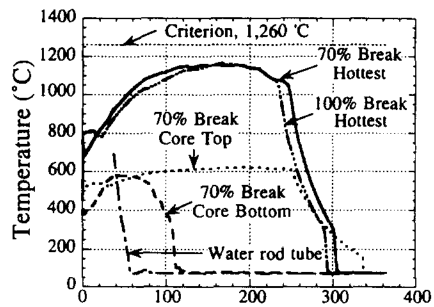

Fig.13 Fuel cladding and water rod tube temperatures in the cold break LOCA

The cold leg break LOCA was analyzed with changing the break size from 40% to 100%. The smaller the break size is, the slower the pressure decreases at the blowdown. The reason why the pressure decreases quickly after 30 sec. is that ADSs are actuated at 30 sec. The LPCIs also starts at 30 sec. when the backup diesel generators start. The ADS are located on the outlet lines and the LPCIs are connected to the main feedwater lines and downcomer and . In the 100% break LOCA, the fuel is efficiently cooled by the blowdown flow. In a smaller size accident, such as 70% LOCA, the cladding surface temperature rises faster. In an accident, such as 40% LOCA, the fuel is efficiently cooled by opening ADS because the core is cooled by the flow towards the ADS vent lines. The 70% break LOCA, hence , gives the most severe cladding temperature. The cladding surface temperature of 70% and 100% cold leg LOCAs are shown in Fig. 13. The maximum cladding surface temperature is 1,150 °C approximately 150 sec. in the 70% LOCA. The criterion for the cladding surface temperature below 1,260 °C is satisfied even if the delay of the refill due to the double tube water rods is considered.

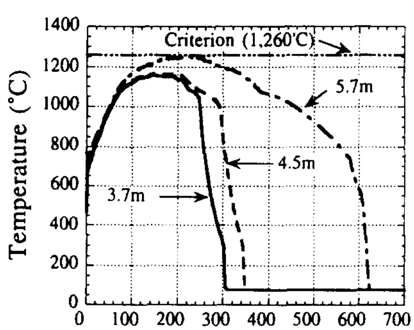

Fig.14 Maximum cladding temperature change with the reactor core height

The effect on the peak clad temperature was studied with changing the core height. The cladding surface temperature of the core height of 3.7 m, 4.5 m and 5.7 m are shown in Fig. 14. The shorter core height causes the faster turn-around time and gives the lower peak clad temperature. This is because two phase flow from the flashing point efficiently cools at the hottest point. From the LOCA analysis, it can be concluded that the core height of 5.7 m is allowable.

7. CONCLUSION

The SCLWR has the once-through coolant system. There is no recirculation line and the coolant inventory is small. The double tube water rods were used for moderation. The coolant density coefficient is approximately twice as much as the of the BWR. The fuel rod diameter is small and lattice pitch is tight. Auxiliary feedwater, automatic depressurization and low pressure coolant injection systems were selected at the safety features. The reactor scram is activated depending on the decrease in the feedwater flow rate. The turbine control and bypass valves and the safety relief valves were actuated succeeding at pressure abnormality.

A computer code for the supercritical pressure, light water cooled reactors, SPRAT, was prepared and safety analysis was carried out. In the loss of flow transient such as the loss of off-site power transient, the core flow rate decreases largely because no natural circulation is established in the once-through type plant. But the reactor power decreases largely because of the coolant density feedback. It is necessary to keep the coast down time of the main feedwater pumps larger than 10 sec. Turbine driven AFWSs with 16% capacity of nominal flow rate are enough to remove the stored and decay heat of the fuel rods and to maintain the core cooling. The increase in coolant flow such as the inadvertent start of the auxiliary feedwater pump increases the power 25% by the coolant density feedback. The overpressurization events such as the generator load rejection is not a severe transient even if scram fails unlike that of the BWR. In the once-through type plant, the decrease in the core flow increases the coolant temperature and coolant density change is small at overpressurization. The reactivity-induced event such as the control rod ejection, is not severe because the reactivity of the control rod is small. Large break LOCA is analyzed. The core heats up quickly because of the small heat capacity of the fuel rods and small coolant inventory in the tight lattice. In addition, the double tube water rods delay the reflood of the core and the peak clad temperature (PCT) is increased. The PCT is 1,150 °C in the 70% cold leg break LOCA. It is below the criterion of the stainless steel clad fuel of 1,260 °C. All analyzed accidents and transients satisfy the criteria.

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