SAFETY OF THE SUPER LWR

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Supercritical water-cooled reactors (SCWRs) are recognized as a Generation IV reactor concept. The Super LWR is a pressure-vessel type thermal spectrum SCWR with downward-flow water rods and is currently under study at the University of Tokyo. This paper reviews Super LWR safety. The fundamental requirement for the Super LWR, which has a once-through coolant cycle, is the core coolant flow rate rather than the coolant inventory. Key safety characteristics of the Super LWR inhere in the design features and have been identified through a series of safety analyses. Although loss-of-flow is the most important abnormality, fuel rod heat-up is mitigated by the "heat sink" and "water source" effects of the water rods. Response of the reactor power against pressurization events is mild due to a small change in the average coolant density and flow stagnation of the once-through coolant cycle. These mild responses against transients and also reactivity feedbacks provide good inherent safety against anticipated-transient-without-scram (ATWS) events without alternative actions. Initiation of an automatic depressurization system provides effective heat removal from the fuel rods. An "in-vessel accumulator" effect of the reactor vessel top dome enhances the fuel rod cooling. This effect enlarges the safety margin for large LOCA.

KEYWORDS : Generation IV Reactor, SCWR, Once-Through Coolant cycle, Supercritical-Pressure, Safety

1. INTRODUCTION

Supercritical water-cooled reactors (SCWRs) are recognized as a Generation IV reactor concept. Several research programs on SCWRs are presently ongoing worldwide [1-4]. The University of Tokyo has been continuously studying the pressure-vessel type SCWR since 1989 [5]. The "Super LWR"is a more recent thermal spectrum reactor design, and is currently being studied at the University of Tokyo. Fig. 1 shows a schematic diagram of the plant system. The Super LWR adopts a once-through coolant cycle operating at supercritical-pressure, similar to a supercritical-pressure fossil-fired power plant (FPP). Since a recirculation system, steam-water separator, steam dryer, steam generator, and pressurizer are not needed, considerable simplification and compactification of the reactor system compared to current light water reactors (LWRs) can be achieved. In addition, compactification of the turbine system and high thermal efficiency are possible, similar to FPPs, due to the high specific heat of the main steam and availability of full-speed steam turbines.

Since the Super LWR is a new reactor concept, its safety characteristics need to be well understood at the concept development phase. To this end, we have proposed a safety principle, designed a safety system, developed a set of computer codes, and analyzed abnormal transients and accidents, including loss of coolant accidents (LOCAs) and anticipated-transient-without-scram (ATWS) [6-9]. It has been found that the Super LWR has several safety characteristics that are unique to the reactor design, i.e., the once-through coolant cycle, supercritical-pressure operation, and downward-flow water rods. This paper summarizes studies on Super LWR safety conducted at the University of Tokyo and comprehensively presents the safety characteristics of this reactor concept.

2. DESIGN FEATURES AND SAFETY PRINCIPLE OF SUPER LWR

Since the safety characteristics of the Super LWR derive from its design, the design features that characterize the reactor behavior under abnormal conditions are summarized first. The coolant cycle is compared with those of LWRs in Fig. 2. One of the fundamental features of the Super LWR is the once-through coolant cycle with inlet pumps and outlet valves; in contrast, LWRs employ a coolant circulation system, i.e., the primary system of a PWR and the recirculation system of a BWR. Another distinctive feature of the Super LWR is single-phase cooling

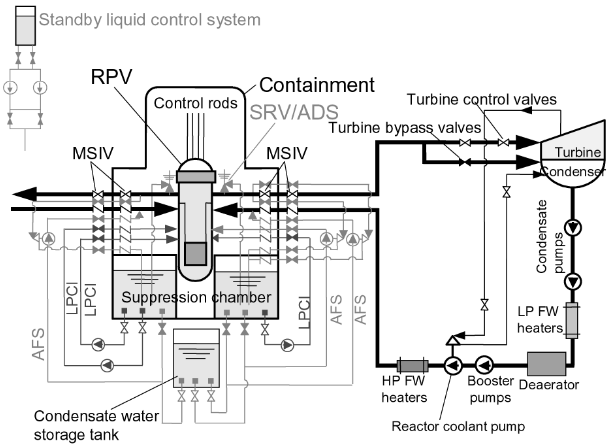

Fig. 1. Plant System of Super LWR

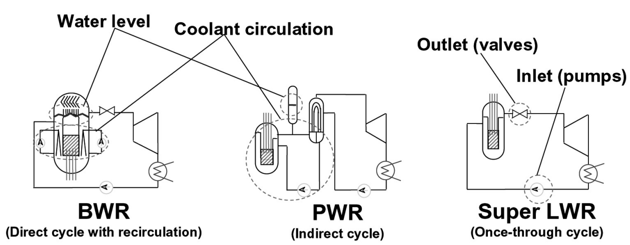

Fig. 2. Comparison of Coolant Cycle

at supercritical-pressure.

In consideration of these design features, the safety principle of the Super LWR was proposed [10]; i.e., to “maintain the core coolant flow”. This is accomplished by maintaining the supply of coolant from the cold leg while also maintaining the discharge of coolant at the hot leg. The corresponding safety principle of LWRs, meanwhile, is to provide a sufficient coolant inventory in order to maintain the reactor vessel water level.

In addition to the coolant cycle described above, design of the fuel assembly (FA) influences the reactor behavior. An example of the plant characteristics is compared with those of a typical BWR and PWR in Table 1. Since the ratio of the core coolant flow rate to the reactor thermal power is about 1/8 of that of a BWR and about 1/9 of that of a PWR, the gap between the fuel rods must be more narrow in order to maintain a high mass flux for fuel rod cooling. Due to the narrow gap and substantial change of the coolant density with axial position, an additional moderator is needed to design a thermal spectrum reactor. The cross-section of the FA is shown in Fig. 3. The Super LWR adopts square water rods [10]. There is heat conduction between the fuel channels and the water rods. The water rods occupy more than 70 % of the total coolant volume in the FA, and thus significantly influence reactor behavior.

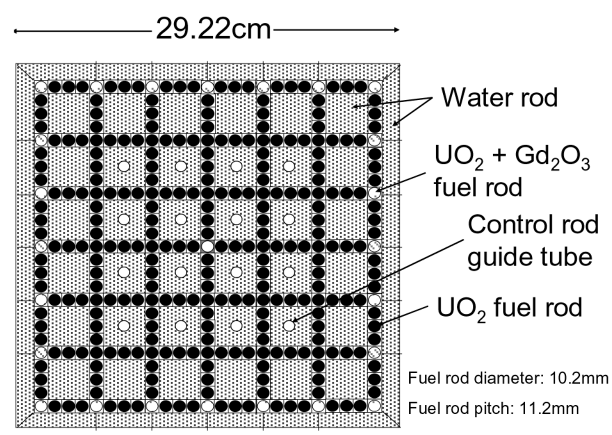

Fig. 3. Cross-Section of a Fuel Assembly

Table 1. Comparison of Typical Plant Characteristics

	Super LWR	BWR	PWR
Pressure (MPa)	25	7.2	15.7
Thermal/Electric power (MW)	2300/1000	3293/1137	3411/1180
Inlet/outlet temperature (°C)	280/500	216/286	289/325
Core coolant / Main steam flow rate	1.19/1.19	13.4/1.78	16.7/1.86
Ratio of core coolant flow rate to thermal power (t/s/GW)	0.52	4.07	4.90

The coolant flow scheme in the reactor pressure vessel (RPV) is shown in Fig. 4. At normal operating conditions, 30% of the coolant entering from the cold legs is directed to the top dome and flows downward through the control rod (CR) guide tubes and the water rods. In the bottom dome, it is mixed with the remaining 70% of the coolant that flows through the downcomer. All the coolant then flows upward through the fuel channels. This flow scheme was proposed from the viewpoint of a steady state design so as to prevent degradation of the average core outlet temperature, provide axially uniform moderation, and separate the high-temperature steam from the pressure boundary [10]. The reactor behavior under abnormal conditions is also influenced by this flow scheme.

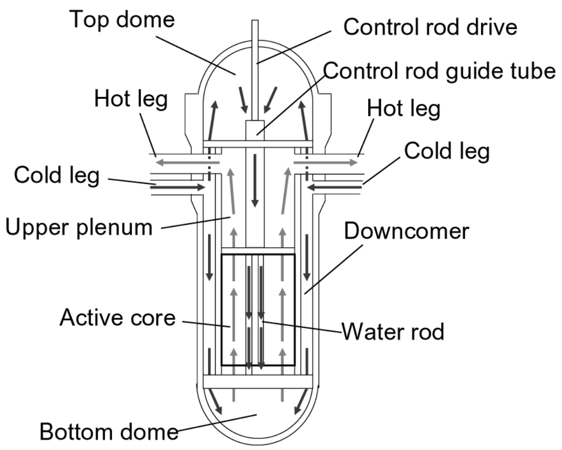

Fig. 4. Coolant Flow Scheme in RPV at Normal Operation

3. SAFETY SYSTEM DESIGN

The safety system of the Super LWR is schematically described in Fig. 1. For reactor shutdown, the reactor trip (scram) system and the standby liquid control system (SLCS) are prepared in the same manner as in a BWR. In consideration of the safety principle, the safety system of the Super LWR needs to have functions for maintaining the supply of coolant from the cold leg and maintaining the discharge of coolant at the hot leg. For the former function, the Super LWR is equipped with a turbine-driven high-pressure auxiliary feedwater system (AFS) and a motor-driven low-pressure core injection (LPCI) system. The AFS plays the role of reactor core isolation cooling (RCIC). The LPCI is one of the functions of the residual heat removal (RHR) system. For the latter function, safety relief valves (SRV) are prepared. The SRV also have the function of acting as an automatic depressurization system (ADS), as in a BWR.

The ADS lends unique behavior to the Super LWR [6,9]. Reactor depressurization by opening 8 ADS valves is analyzed using the SPRAT-DOWN-DP code, introduced in Chapter 4.2. Coolant flow during reactor depressurization is shown in Fig. 5. The supply of coolant from the cold leg is assumed to stop in 5 s. The analysis result is shown in Fig. 6. Initiating the ADS induces strong core coolant flow. This safety characteristic derives from the once-through coolant cycle without a recirculation system.

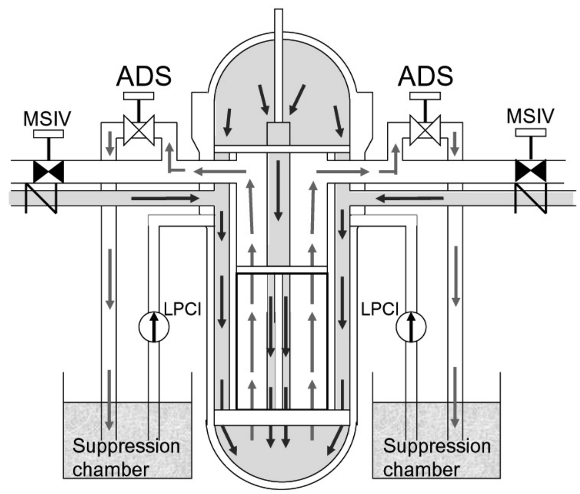

Fig. 5. Coolant Flow During Depressurization

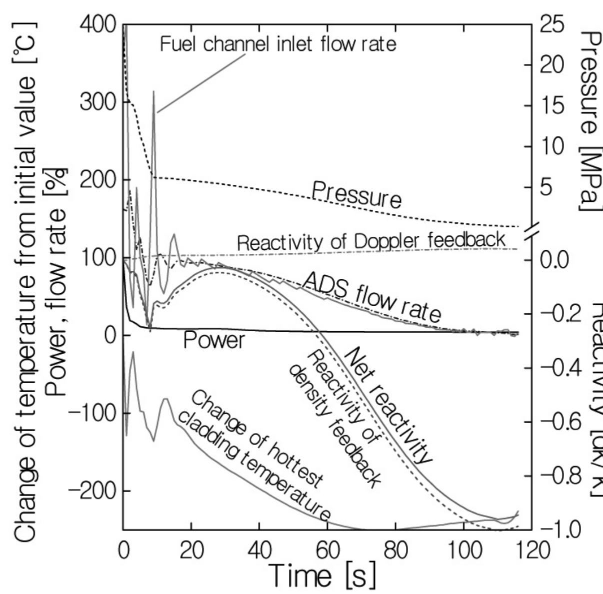

Fig. 6. Depressurization Behavior

During depressurization, the top dome passively supplies its coolant inventory to the fuel channels in a manner similar to an “in-vessel accumulator”. This is a key advantage of a core with downward-flow water rods, because the downward-flow water rod is not a bypass flow path. The core coolant flow rate is maintained even when the supply of coolant from the cold leg has stopped. Also, depressurization decreases the reactivity, because the Super LWR has a negative void reactivity coefficient. Due to these thermal-hydraulic and neutronic effects of reactor depressurization, the hottest cladding temperature does not exceed the initial value. Discharge of the coolant inventory does not threaten safety, because maintaining the coolant inventory is not the fundamental safety requirement for the once-through coolant cycle as long as the core coolant flow is maintained. After depressurization, the core is cooled by the LPCI.

The principle for actuating the safety system was proposed [6], and is summarized in Table 2. Abnormalities in supplying coolant from the cold leg are detected as “flow rate low” levels, while abnormalities in discharging coolant at the hot leg are detected as “pressure high” levels. When the decay heat cannot be removed at supercritical-pressure, which corresponds to a low level 3 flow rate, the reactor is depressurized, as illustrated in Fig. 6, and then cooled by the LPCI. When the pressure decreases from the supercritical to subcritical region, boiling transition will occur on the fuel rod surface, leading to a rapid increase in the cladding temperature. It is known that the minimum heat transfer coefficient is especially small just below the critical pressure. Therefore, the pressure should NOT stay or decrease slowly around the critical pressure. The reactor is depressurized at a low level 2 pressure, which is 106 % of the critical pressure.

The safety system design is summarized in Table 3. The capacities and actuation conditions were determined in reference to LWRs and also considering the uniqueness of the Super LWR [6]. For example, the ADS and the main steam isolation valve (MSIV) are actuated by the same signal, because closing the MSIV without initiating the ADS causes flow stagnation in the once-through coolant cycle without recirculation.

Table 3. Summary of Safety System Design

<table>
<tr><th>Safety system</th><th colspan="5">Actuation conditions</th></tr>
<tr><td>Reactor trip (scram) system</td><td colspan="5">Pressure high (level 1)
Pressure low (level 1)
MSIV closure (90%)
ECCS start-up
Reactor power high (120%)
Drywell pressure high
Reactor coolant pump trip
Loss of offsite power
Condensate pump trip
Main coolant flow rate low (level 1)
Turbine control valve quickly closed
Main stop valve closure
Reactor period short (10s)
Earthquake acceleration large</td></tr>
<tr><td>AFS
(4% of rated flow ×3 units, Turbine driven)</td><td colspan="5">Reactor coolant pump trip
Loss of offsite power
Condensate pump trip
Main coolant flow rate low (level 2)
Turbine control valves quickly closed
Main stop valves closure
MSIV closure (90%)</td></tr>
<tr><td rowspan="6">SRV
(20% of rated flow × 8 valves)</td><td colspan="3">Relief valve function</td><td colspan="2">Safety valve function</td></tr>
<tr><td>Open (MPa)</td><td>Close (MPa)</td><td>Number</td><td>Open (MPa)</td><td>Number</td></tr>
<tr><td>26.2</td><td>25.2</td><td>1</td><td>27.0</td><td>2</td></tr>
<tr><td>26.4</td><td>25.4</td><td>1</td><td>27.2</td><td>3</td></tr>
<tr><td>26.6</td><td>25.6</td><td>3</td><td>27.4</td><td>3</td></tr>
<tr><td>26.8</td><td>25.8</td><td>3</td><td></td><td></td></tr>
<tr><td>ADS
(One of the SRV functions)</td><td colspan="5" rowspan="3">Pressure low (level 2)
Main coolant flow rate low (level 3)
Drywell pressure high</td></tr>
<tr><td>MSIV</td></tr>
<tr><td>LPCI
(12% of rated flow ×3 units, Motor driven)</td></tr>
</table>

Table 2. Principle of Actuating Safety System

Flow rate low		
	Level 1	Reactor scram
	Level 2	AFS
	Level 3	ADS/LPCI
Pressure high		
	Level 1	Reactor scram
	Level 2	SRV
Pressure low		
	Level 1	Reactor scram
	Level 2	ADS/LPCI

4. SAFETY ANALYSIS CODES

The safety analysis codes for LWRs cannot be applied to the Super LWR without major modifications, because the operating pressure, the flow scheme, and the fuel bundle geometry are all different. The purpose of safety analyses here is to clarify the characteristics of the reactor behavior in the concept development phase. The calculation models and the geometries must be flexibly changed with the reactor design. Therefore, simple codes using a 1-D node-junction model are suitable for studying the Super LWR in this phase. Three codes were developed for modeling the supercritical-pressure condition, reactor depressurization, and core reflooding, respectively [7-9,11].

4.1 Analysis Code for Supercritical-Pressure Condition

The analysis code for supercritical-pressure condition is called SPRAT-DOWN. The nodalization is shown in Fig. 7. Mass and energy conservations are solved. Flow redistribution between the parallel flow paths is considered along with calculation of the pressure drop balance and momentum conservation.

The peak cladding temperature (PCT) is limited for fuel rod integrity at abnormal transients, accidents, and ATWS, as described in Chapter 6.1. The heat transfer coefficient under the limiting condition is important from a safety viewpoint. Loss-of-flow events give the highest PCTs for each category.[1)] These PCTs appear under the thermal-hydraulic condition of high bulk temperature (above 600°C for transients and above 800°C for accidents and ATWS) and low mass flux (below 600 kg/m^{2}s for transients and below 200kg/m^{2}s for accidents and ATWS). The Dittus-Boelter correlation is a good benchmark at such high bulk temperatures. Oka-Koshizuka's correlation, also called Kitoh's correlation, was developed based on single-phase numerical simulations using Jones-Launder's k-ε turbulence model [12,13]. This correlation gives slightly lower heat transfer coefficients than the Dittus-Boelter correlation under the highest PCT conditions. For the purposes of this study, the accuracy requirement for the heat transfer coefficient is not strict, because there is a margin between the highest PCTs and the criteria. The Oka-Koshizuka correlation is applied to the SPRAT-DOWN. Sensitivity of the heat transfer coefficient in the case of the highest PCTs was found to be small [7-9].

The point kinetics model is used for neutronics calculation. The point-kinetics model is a good approximation to analyze the transient behavior as long as the reactivity does not change locally and the space effect on the reactivity feedback is considered. It is still a good approximation even for localized CR withdrawal events as long as the power changes slowly over a longer time scale relative to the effective neutron lifetime. Since the Super LWR is a water-cooled thermal spectrum reactor, like LWRs, where the coolant density feedback and Doppler feedback are

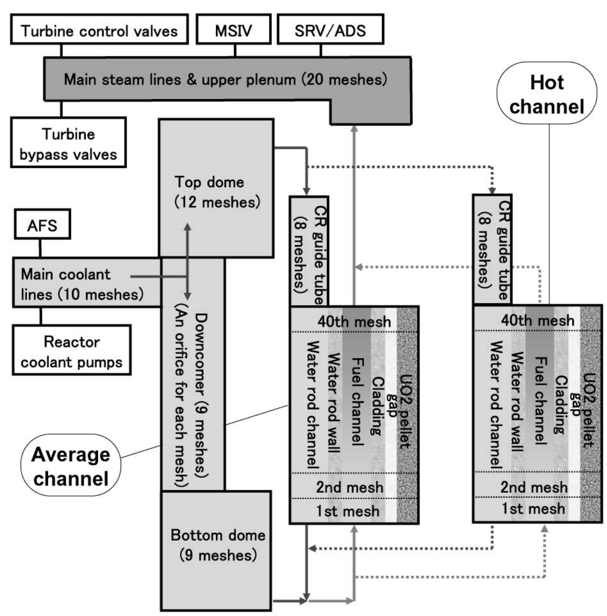

Fig. 7. Nodalization of SPRAT-DOWN

[1)] Small LOCA gives the highest PCT of accidents. This is a kind of loss-of-flow event for the Super LWR.

dominant, only these feedbacks are considered. The density coefficient and the Doppler coefficient were determined from the 3-D core design [10]. The space effect on reactivity feedback is considered by calculating the "average" values of the coolant density and pellet temperature at each time step. Contribution of each mesh to the "average" values is proportional to the square of the linear power density (chopped-cosine distribution). Decay heat is calculated with a two-group approximation of the "ANS+20%" model.

4.2 Blowdown Analysis Code

The blowdown analysis code was developed based on SPRAT-DOWN [4], and is called SPRAT-DOWN-DP. The nodalization is the same as that of SPRAT-DOWN. In the mass and energy conservation calculation, the homogeneous equilibrium model (HEM) is applied to two-phase meshes. Flow redistribution between parallel paths is not calculated in contrast with SPRAT-DOWN. Instead, flow boundary conditions are used as shown in Fig. 8 [8,9]. The effect of the flow boundary condition on the PCT was investigated [9]. Since there is no fuel rod in the water rod path or the downcomer path, the flow redistribution between these paths does not influence the PCT. The hot /average channels are not distinguished, unlike in SPRAT-DOWN. The influence of the flow rate ratio between the hot / average channels was also investigated [9]. Although the hot channel flow rate is conservatively assumed to be half of the average channel flow rate, the increase in the PCT is about 300°C, which is sufficiently smaller than the safety margin.

The radial heat transfer model is the same as that of SPRAT-DOWN. During depressurization, the PCT

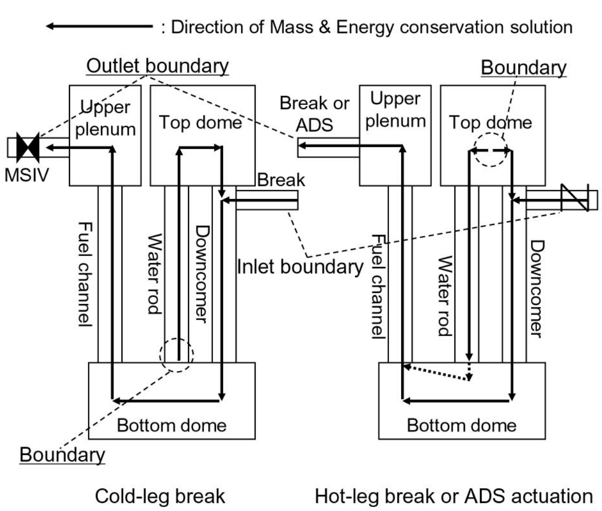

Fig. 8. Boundary Conditions of SPRAT-DOWN-DP

appears at a superheated-steam condition. The heat transfer coefficient of this condition is important. The Dittus-Boelter correlation, which is widely used for LWRs, is applied. The heat transfer coefficient at the two-phase condition is less important because the cladding temperature is always lower than the hottest cladding temperature at the normal operating conditions. Dougal-Rhosenow's film boiling correlation is conservatively applied to both pre-CHF and post-CHF conditions.

Change of reactivity is dominated by coolant density feedback and Doppler feedback during depressurization. Since these feedbacks are not localized, the reactivity and the reactor power are calculated with the same model as that of SPRAT-DOWN.

SPTAT-DOWN-DP was compared with the REFLA-TRAC code developed by Japan Atomic Energy Agency (JAEA) as a best-estimate code for LWRs [8]. It was shown that SPRAT-DOWN-DP is applicable to the Super LWR for the purpose of concept development.

4.3 Reflooding Analysis Code

For a reflooding analysis, the "SCRELA reflood estimation module" was developed [11]. It includes the "system momentum calculation", the "thermal equilibrium relative velocity correlation" and the "quench front velocity correlation". Various heat transfer correlations are prepared according to the flow conditions, such as single-phase liquid, saturated two-phase, transient, dispersed, and superheated steam flow.

Applicability of this code to the tight lattice bundle (P/D$\approx$1.1 for the Super LWR) was assessed on the basis of comparison with the NEPTUN LWHCR experiment (P/D$\approx$1.13) [8]. SCRELA predicted the quench front propagation to be slightly slower than that of the experiment. The cladding temperature was calculated to be higher than that of the experiment. It was concluded that SCRELA could be applied to the Super LWR in the concept development phase.

The water rods are neglected in SCRELA. This can be considered conservative from viewpoint of a heat sink. The effect of the water rods on the quench front propagation needs to be assessed in future study.

5. EVENT SELECTION AND CLASSIFICATION

Since the Super LWR is a kind of simplified light water reactor, its abnormal events were selected from those of LWRs [7]. They are summarized in Table 4 together with those of a PWR and a BWR.

The abnormal events related to a "decrease in core coolant flow rate" are the most important for the Super LWR, because the core coolant flow rate is a fundamental safety requirement, as described in Chapter 2. Since the coolant cycle of the Super LWR is different from that of

Table 4. Comparison of Abnormal Events - O: transient, ◆: accident – (Some Events are Described Repeatedly.)

Type of abnormality	Event	PWR	BWR	Super LWR
Abnormality in reactivity and power distribution	Uncontrolled CR withdrawal	O	O	O
	CR assembly misalignment and drop	O		O
	CR ejection	◆		◆
	CR drop		◆	
	Boron dilution	O		
	Start-up of an inactive reactor coolant loop	O	O	(1)
	Loss of feedwater heating		O	O
	Reactor coolant flow control system failure		O	O
	Feedwater control system failure		O	
	Inadvertent start-up of AFS			O
Decrease in core coolant flow rate	Partial loss of reactor coolant flow	O	O	O
	Total loss of reactor coolant flow	◆ (2)	◆	◆
	Loss of offsite power	O	O	O
	Loss of turbine load			O
	Isolation of main steam line			O
	Reactor coolant pump seizure	◆	◆	◆
Abnormality in reactor pressure and coolant inventory	Loss of offsite power		O	O
	Loss of turbine load		O	O
	Isolation of main steam line		O	O
	Depressurization of reactor coolant system	O		O
	Inadvertent start-up of ECCS	O	O	
	Pressure control system failure		O	O
	Inadvertent SRV opening		O	O
	Loss of all feedwater flow		O	(3)
	LOCA	◆	◆	◆
	Main steam line break	◆	(4)	(4)
	Main feedwater pipe rupture	◆	(4)	(4)
Abnormality in secondary system	Loss of turbine load	O		
	Load increase	O		
	Depressurization	O		
	Loss of all feedwater flow	O		
	Over SG water feed	O		
Abnormality in containment	LOCA	◆	◆	◆
	Generation of H2 gas	◆	◆	◆
	Dynamic load to containment		◆	◆
Radioactive release	Waste gas decay tank rupture	◆	◆	◆
	Improper fuel assembly insertion or drop	◆	◆	◆
	Main steam line break outside containment		◆	◆
	SG tube rupture	◆		
	LOCA	◆	◆	◆
	CR ejection	◆		◆
	CR drop		◆	

(1) not possible for the Super LWR due to two loop configuration
(2) classified as an accident for Japanese PWRs
(3) the same as "Total loss of reactor coolant flow" due to absence of recirculation or secondary system
(4) covered by LOCA due to absence of secondary system

both a PWR and a BWR, the events need to be carefully selected and classified. The once-through coolant cycle of the Super LWR is schematically illustrated in Fig. 9. The feedwater pump is the same as the reactor coolant pump (RCP). A "loss of all feedwater flow" and a "total loss of reactor coolant flow" are the same incidents. Classification of this event depends on the frequency. We followed the guidelines for Japanese LWRs. A simultaneous sudden trip of all pumps that have been directly maintaining the core coolant flow rate is classified as a "total loss of reactor coolant flow" accident. These pumps correspond to the primary pumps of a PWR and the recirculation pumps of a BWR. Since the RCPs of the Super LWR also maintain the core coolant flow rate, a simultaneous sudden trip of the RCPs is classified as a "total loss of reactor coolant flow" accident, assuming that its frequency will be less than 10^{-3} per year by system separation and high reliability.

Loss of supply of coolant to the deaerator would also cause a trip of the RCPs, because the RCP inlet pressure decreases with the deaerator water level. This abnormality is represented by a "loss of offsite power" transient where the motor-driven condensate pumps stop. Since there is a large amount of water in the deaerator, the RCPs are expected NOT to stop for some period after the trip of the condensate pumps. The capacity of the deaerator has not yet been determined. If it is 140 m^3, which corresponds with the typical design of a 1000 MWe class FPP, the water level in the deaerator would decrease by only 7% in 10 s after the trip of the condensate pumps. In the safety analysis, it was conservatively assumed that the trip of the RCPs occurs 10s after the condensate pump trip [7]. This transient is less severe than a "total loss of reactor coolant flow" accident, because a reactor scram is possible before the trip of the RCPs. In the safety analysis, the reactor scram by the signal of "loss of offsite power" or "condensate pump trip" or "turbine control valves quickly closed" was credited.

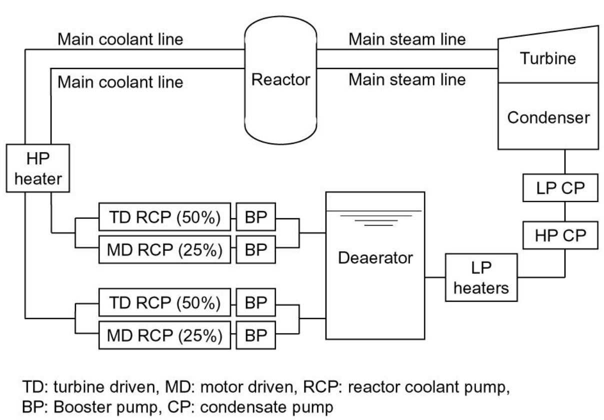

Fig. 9. Schematic Diagram of a Once-Through Coolant Cycle

Loss of the supply of steam to the turbine-driven RCPs would also cause a trip of the RCPs. A "loss of turbine load" transient and an "isolation of main steam line" transient are followed by this situation. The RCPs are assumed NOT to stop for some period, because there is residual steam in the main steam lines and the turbines, similar to a BWR. In the safety analysis, the trip of the RCPs is assumed to occur 10 s after the initiation of the transients. A reactor scram by the signal of "turbine control valves quickly closed" or "MSIV closure" was credited before the trip of the RCPs.

The event selection and classification for other types of abnormalities are not unique to the Super LWR. Each event was selected from a PWR or a BWR in consideration of similarities in the plant system. For example, the abnormalities in the CR were taken from those of a PWR, because of the cluster type CR inserted from the core top, similar to a PWR. Pressurization transients such as a "loss of turbine load", "isolation of main steam line", etc were taken from those of a BWR, because of the direct cycle, corresponding to a BWR.

6. SAFETY CRITERIA

Since the Super LWR is presently in the concept development phase, the safety criteria cannot be determined based on experiments. We determined the principle for the safety criteria and the tentative values for the safety analyses [7-9]. The requirements for abnormal transients are the same as those of LWRs: no systematic fuel rod damage and no pressure boundary damage. The requirement for accidents is no excessive core damage, which also corresponds with LWRs. The safety criteria described below are determined for concept development. Experiments will be necessary for assessing their validity.

6.1 Criteria for Fuel Rod Integrity

The fuel rod cladding material of the SCWR is under screening and development [14]. For fuel rod design of the Super LWR in the concept development phase, typical austenitic stainless steels or Ni-alloys are applied [15-17]. The principle of the safety criteria for fuel rod integrity is shown in Table 5. Since heat transfer deterioration is a much milder phenomenon than boiling transition, the minimum deterioration heat flux ratio was eliminated from the transient criterion related to fuel rod heat-up [12].

We separated the types of abnormalities into "loss of cooling" and "overpower". For the "loss of cooing" type transients, the limiting failure mode is expected to be buckling collapse of the cladding due to a decrease in the Young's modulus at high temperature. Yamaji, et al. determined the maximum allowable temperature of the cladding as 850 °C, taking several conservatisms, so that

Table 5. Principle of Safety Criteria for Fuel Rod Integrity

Category	Requirement	Mechanical failure			Heat-up
		Buckling	Burst	PCI	
Accident	No excessive damage			Enthalpy<Limit (RIA)	Oxidation<Limit MSCT<Limit
Transient	No systematic damaage	ΔP on Clad. <Limit	Plastic strain <Limit	Pellet temp.<Limit Plastic strain <Limit Enthalpy<Limit	

: Loss of cooling
: overpower

the pressure difference on the cladding is less than one-third of the collapse pressure [17].

For "loss of cooling" type accidents, the requirement is to maintain a coolable geometry, as in LWRs. The limiting failure mode is expected to be oxidation of the cladding. The criterion of the cladding temperature is set at 1260 °C for stainless steels, taken from the criterion for LOCA of US PWRs with stainless steal cladding [18]. The criterion for Ni-alloys is also set at 1260 °C, because the neutronic composition of Ni-alloys is similar to that of stainless steels.

For "overpower" type transients, the limiting failure mode is expected to be burst or PCI. Yamaji, et al., determined the maximum allowable power using the FEMAXI-6, a fuel rod analysis code for LWRs developed by JAEA, taking several conservatisms so as to prevent melting of the pellet centerline and plastic strain of the cladding [17]. When the power rise rate is small (0.1 – 1 % of the initial power per second), the allowable power is 124% of the rated power. When it is relatively large (1 – 10 %), the allowable power is 136%. When it is larger than 10%, the allowable power is 182%. A transient with reactivity insertion over $1 is not expected in the Super LWR, because the reactor is scrammed before a CR cluster is fully withdrawn. Thus, the maximum allowable fuel enthalpy is not taken as a criterion for abnormal transients.

For "overpower" type accidents such as CR ejections, the maximum allowable fuel enthalpy needs to be determined, as in LWRs. In this phase, the same criterion as that of LWRs (230 cal/g) is taken. The validity should be assessed by experiments in the future.

6.2 Criteria for Pressure Boundary Integrity

The relative pressure change of the Super LWR is smaller than that of LWRs due to the once-through coolant cycle and the high operating pressure [7,9]. The maximum allowable pressures for transients and accidents were set at 105 % and 110 % of the maximum pressure of normal operation, respectively, while those of LWRs are 110 % and 120 %. Since the average core outlet temperature is high, similar to LMFBRs, thermal creep of the main steam lines should be considered for both normal and abnormal conditions in future designs. It might be reasonable to limit the duration of high temperature by considering the cumulative damage fraction (CDF), as is done for LMFBRs. Material of the main steam lines used in FPPs, such as 9Cr-1Mo etc, has high creep strength and is a promising candidate material, because the operating temperature and pressure of the Super LWR are within those of recently constructed FPPs.

6.3 Criteria for ATWS

An ATWS is defined as an abnormal transient followed by failure of a reactor scram. Since the Super LWR is a simplified light water reactor, the probability of an ATWS is expected to be on the same order as that of LWRs. An ATWS of the Super LWR is classified as a "beyond design basis event" (BDBE). A deterministic evaluation of an ATWS is a global requirement, because it is a potential safety issue that may lead to core damage under postulated conditions. Also, it is expected that inherent safety characteristics of nuclear reactors, not only reactivity feedback but also reactor system dynamics, can be clearly identified at ATWS conditions due to a scram failure. Therefore, deterministic ATWS analyses were carried out for the Super LWR [9].

Despite the significantly low probability of an ATWS compared to other accidents, the same criteria as those of the accidents were applied, as in LWRs. It should be noted that fission-product (FP) gas release to the coolant does not bring about an issue of positive reactivity insertion, because the Super LWR has a negative void reactivity coefficient, as in LWRs.

7. SAFETY ANALYSES

7.1 Initial Conditions

The plant characteristics written in Table 1 are used for the safety analyses. The initial conditions are shown in Table 6. The hottest cladding temperature of 650°C is the same as the criterion applied in the three-dimensional core design where single-channel thermal hydraulic analyses are carried out for homogenized fuel assembles [6]. However, the cladding temperature is expected to be higher due to pin-by-pin power distribution and subchannel-by-subchannel flow distribution. The increase in the hottest cladding temperature by these local effects was evaluated to be about 60 °C using subchannel analyses coupled with fuel assembly burnup calculations [19]. The increase in the hottest cladding temperature by engineering uncertainties was evaluated to be about 30 °C using a Monte Carlo statistical thermal design procedure [20]. In consideration of these temperature increases, the allowable increase in the hottest cladding temperature from the initial conditions is set to 110 °C (850°C – [650°C + 60°C + 30°C]) for transients, and 520 °C (1260°C – [650°C + 60°C + 30°C]) for accidents and ATWS.

Table 6. Initial Conditions for Safety Analyses (Hot channel is treated as average channel in LOCA analyses.)

Fuel channel	Average	Hot
Maximum linear power (kW/m)	28	39
Mass flux (kg/s/m^2)	945	1159
Coolant inlet/outlet temperature (°C)	305/500	305/573
Hottest cladding temperature (°C)	-	650
Water rod channel	**Average**	**Hot**
Mass flux (kg/s/m^2)	94	113
Coolant inlet/outlet temperature (°C)	280/366	280/348

Table 7. Initiating Events of Safety Analyses

Transients
Decrease in core coolant flow rate
1 Partial loss of reactor coolant flow
2 Loss of offsite power
Abnormality in reactor pressure
3 Loss of turbine load
4 Isolation of main steam line
5 Pressure control system failure
Abnormality in reactivity
6 Loss of feedwater heating*
7 Inadvertent startup of AFS*
8 Reactor coolant flow control system failure
9 Uncontrolled CR withdrawal at normal operation
10 Uncontrolled CR withdrawal at startup
Accidents
Decrease in core coolant flow rate
1 Total loss of reactor coolant flow
2 Reactor coolant pump seizure
Abnormality in reactivity
3 CR ejection at full power
4 CR ejection at hot standby
LOCA
5 Large LOCA
6 Small LOCA

*ATWS is not analyzed because the scram condition does not occur.

7.2 Safety Analyses Results

The limiting events for each type of abnormality were selected from Table 4 for the safety analyses and are summarized in Table 7. The safety characteristics of the Super LWR are presented here using typical results.

7.2.1 Decrease in Core Coolant Flow Rate

In the case of "total loss of reactor coolant flow" (accident No.1 in Table 7), the main coolant flow rate is assumed to decrease linearly in 5 s due to a simultaneous sudden trip of the RCPs and to remain at zero until initiation of the AFS at 30 s. The analysis results are shown in Fig. 10. The power decreases to the decay heat level due to a reactor scram. Reverse flow occurs in the water rod channel, because the buoyancy pressure drop dominates the pressure drop balance. When the coolant temperature in the fuel channel increases, heat conduction to the water rods also increases. This implies that the water rods serve as a "heat sink". As the coolant expands in the water rods due to heat-up, there is an increase in the flow rate at the downstream of the water rods, including the fuel channel inlet. Consequently, the fuel channel flow rate is maintained even though the coolant supply from the cold leg has stopped. This is called the "water source" effect of the water rods. The "heat sink" and "water source" effects mitigate heat-up

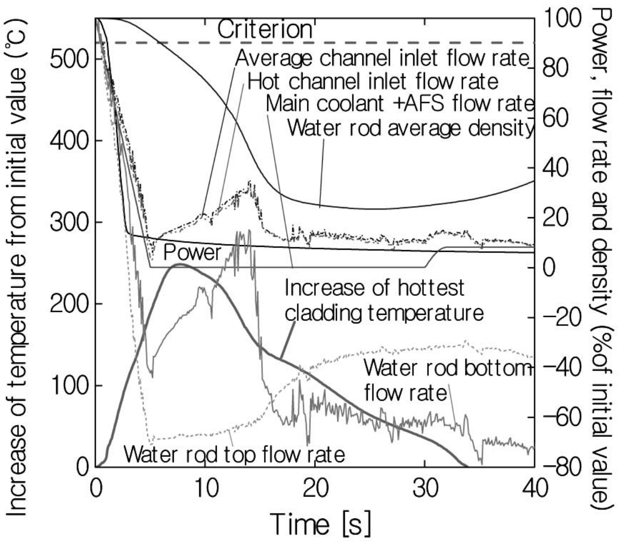

Fig. 10. Total Loss of Reactor Coolant Flow Accident

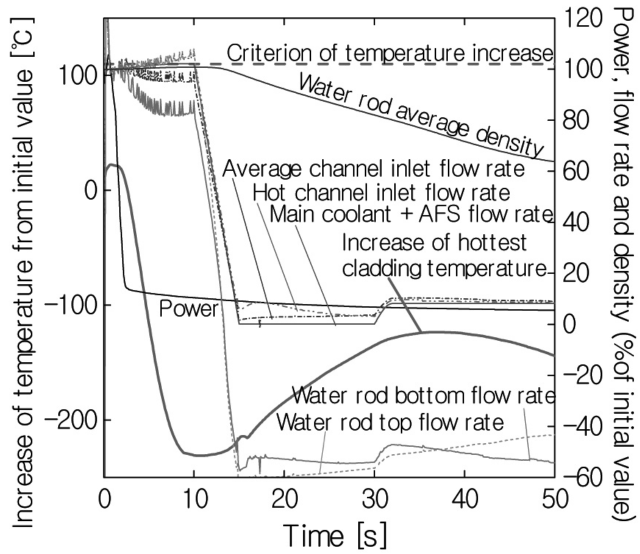

Fig. 11. Loss of Offsite Power Transient

of the fuel rod cladding. The hottest cladding temperature begins to decrease before initiation of the AFS.

At a “loss of offsite power” (transient No.2), the RCPs are assumed to trip at 10 s, as described in Chapter 5. The analysis results are shown in Fig. 11. Since the reactor is scrammed before the trip of the RCPs, the PCT does not exceed the initial temperature.

When the reactor scram fails at this transient, it becomes the limiting ATWS event of the Super LWR. The analysis results without alternative action are shown in Fig. 12. The reactor power decreases to the decay heat level after the trip of the RCPs due to the density feedback. The fuel channel flow rate is maintained by the “water source” effect of the water rods. Therefore, the cladding temperature begins to decrease at 17s. After low temperature coolant from the AFS enters the water rods, however, the reactor returns to criticality. The power remains higher than the decay heat level. After 43 s, the cladding temperature increases again and continues increasing until 177 s. The second peak of the hottest cladding temperature is higher than the first peak. However, the criterion is still satisfied with margin. The reactor has almost reached a high temperature stable condition by 700 s.

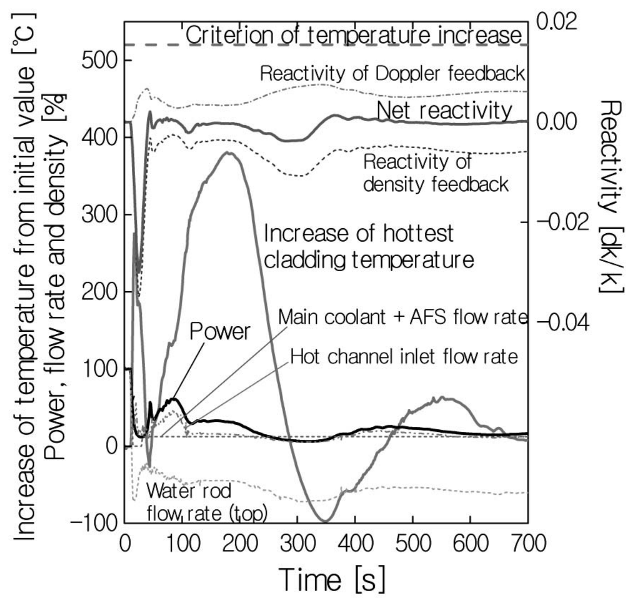

Fig. 12. Loss of Offsite Power ATWS Without Alternative Action

7.2.2 Abnormality in Reactor Pressure

As a typical pressurization event, “loss of turbine load” (transient No.3) is presented here. The turbine bypass is assumed to fail. A reactor scram is not credited, and thus it is considered an ATWS event. The analysis results are shown in Fig. 13. The pressure increases due to closure of the turbine control valves without a turbine bypass. The average coolant density is less sensitive to the pressure compared to a BWR due to the absence of a void collapse and a smaller density difference between “steam” and “water”.[2] Closure of the coolant outlet of the once-through coolant cycle causes flow stagnation in the core, which suppresses the increase in the coolant

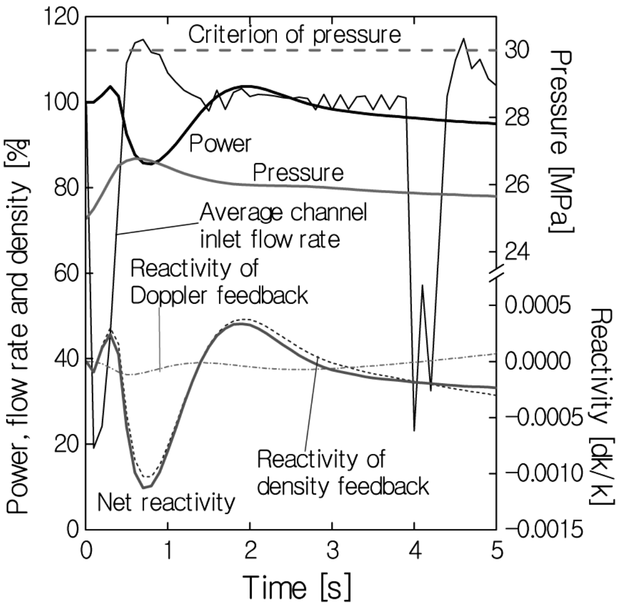

Fig. 13. Loss of Turbine Load ATWS

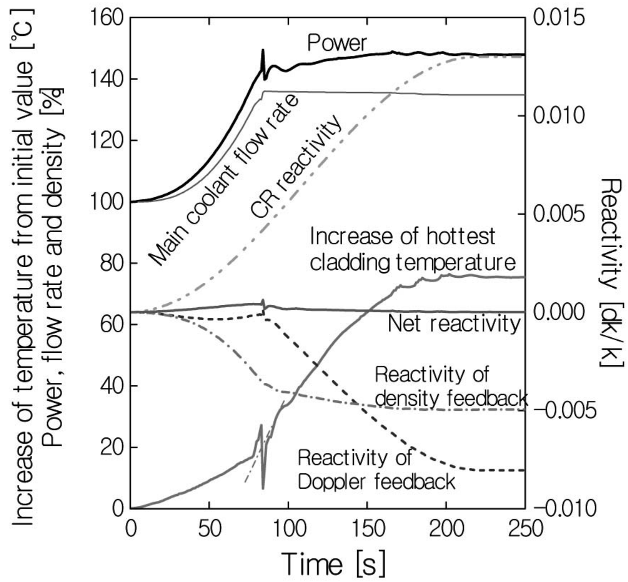

Fig. 14. Uncontrolled CR Withdrawal at Normal Operation ATWS

density due to an increase in the temperature. These inherent characteristics of the Super LWR make the reactivity insertion and power increase significantly small. When the SRVs are opened, the pressure begins to decrease. The peak pressure is much lower than the criterion without alternative action. The pressurization events also lead to loss-of-flow events, because the steam supply to the turbine-driven RCPs would stop, as discussed in Chapter 5. The reactor behavior after the trip of the RCPs at 10 s is almost the same as that described in Fig. 11 or Fig. 12 with/without a reactor scram, respectively.

7.2.3 Abnormality in Reactivity

"Uncontrolled CR withdrawal at normal operation" (transient No. 9) without a reactor scram is presented here. Although the CR withdrawal itself would be stopped at a certain power level by an interlock system independent of the reactor trip system, a CR cluster having the maximum reactivity worth in this reactor is conservatively assumed to be fully withdrawn. It is assumed to be 1.3 %dk/k, which was determined with a very conservative assumption [7]. The same withdrawal speed as that of a PWR (114 cm/min) is assumed. The calculation results of begin-of-cycle (BOC) are shown in Fig. 14. The rate of power increase is small because of reactivity feedbacks. The main coolant flow rate increases with the reactor power due to operation of the main steam temperature control system, which was designed in a previous study [21]. The reactor has almost settled to a high-temperature stable condition by 250 s. All the criteria for ATWS are well satisfied. When the main steam temperature control system is not considered, the power/flow mismatch worsens. This results in a higher cladding temperature. The PCT is higher than the reference case by 140 °C. This is also well below the criterion.

Reactivity abnormalities caused by an increase in the coolant density (transient No.6-8) are mild. They are settled by the power control system in the cases of No.6,7 or a reactor scram in the case of No.8 or Doppler feedback in the case of No.8 as an ATWS event.

7.2.4 LOCA

A large LOCA of the Super LWR is defined as a pipe break accident where the core pressure decreases to the ADS setpoint (23.5MPa) even if the RCPs and the pressure control system are assumed to operate [8]. A large LOCA of the Super LWR is a 15 - 100 % cold leg break or a 34 -100 % hot leg break.[3] These ranges depend on the pipe diameters and initial temperatures.

A 100% cold leg break is presented here as a typical large LOCA. The analysis results of the blowdown phase

[2] Steam/water means above/below the pseudo-critical temperature for supercritical-pressure.

[3] A 100% break is the largest due to a "single-ended break" of the once-through coolant cycle [8].

are shown in Fig. 15. Initially, the cladding temperature increases, because the discharge of coolant from the cold leg decreases the core coolant flow rate. However, the ADS are initiated within 1 s by detecting the "pressure low level 2" and the core coolant flow is subsequently recovered. The increase in the hottest cladding temperature is much smaller than the criterion. This is not highly sensitive to the break ratio but sensitive to the ADS delay time and the number of actuated ADS valves [8]. The cladding temperature is maintained at a low level during blowdown due to the "in-vessel accumulator" effect described in Chapter 3. It is below the initial value when the core reflooding starts. The analysis results of the reflooding phase are shown in Fig. 16. Since the initial temperature of the reflooding phase is low, the PCT is much lower than the criterion. It is not highly sensitive to the LPCI capacity or the axial power distribution. However, the reflooding speed and the PCT are considerably sensitive to the submergence of the quencher in the suppression pool due to the pressure drop from head loss [8]. It implies that the suppression pool and the quencher should be carefully designed.

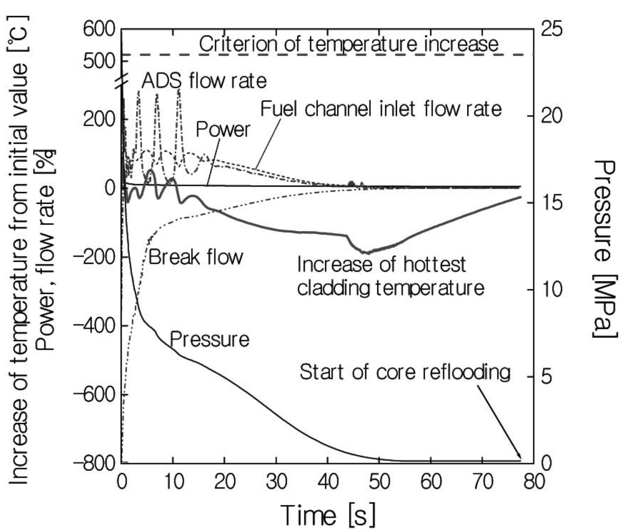

Fig. 15. Blowdown Phase of 100% Cold Leg Dreak LOCA

The small LOCA is defined as a pipe break accident where the core pressure stays above the ADS setpoint, and includes the case where the pressure control system is considered. Since the pressure stays at the supercritical region, a cold leg break is a kind of flow-decreasing event. When the break ratio is at the upper limit of the small LOCA, the core coolant flow rate is the smallest and the increase in the hottest cladding temperature is

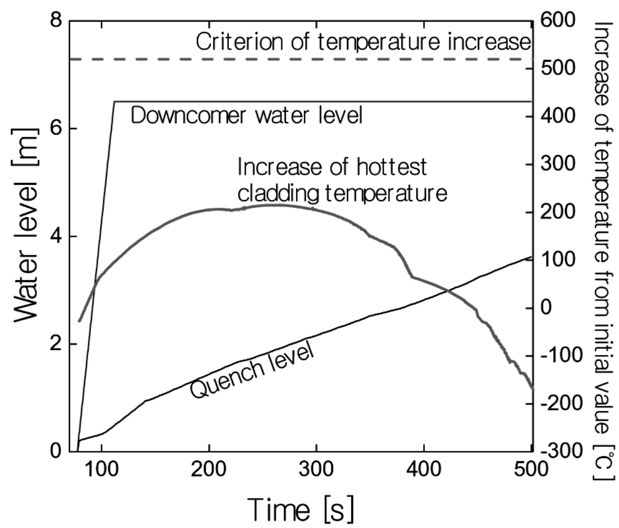

Fig. 16. Reflooding Phase of 100% Cold Leg Break LOCA

about 360 °C, which is higher than that in a large LOCA but is still below the criterion of 520°C. If ADS signals other than the low-pressure setpoint are considered, such as "drywell pressure high" or "mismatch of main coolant / main steam flow rates", the highest temperature of the small LOCA is within that of a large LOCA.

7.3 Summary of Safety Analyses and Discussion of Safety Characteristics

The increases in the hottest cladding temperature from the initial value are summarized in Fig. 17 for events with fuel rod heat-up. The key safety characteristics at loss-of-flow type events are that the temperature increase is small at the transients (No.2-4) due to the reactor scram before the trip of the RCPs, and that excessive heat-up of the fuel cladding is suppressed at the accidents (No.1,5) and ATWS (No.2-4) by the "heat sink" and "water source" effects of the water rods. Opening the ADS valves increases the core coolant flow rate and the top dome passively supplies its coolant inventory to the fuel channels in the manner of an "in-vessel accumulator". These are key advantages of a core with a once-through coolant cycle and downward-flow water rods. Opening the ADS valves suppresses excessive heat-up of the fuel rod at large LOCA. At abnormal transients, the duration of the high cladding surface temperature is very short, as shown in Table 8. This can be considered in evaluating the CDF in future R&D.

The peak powers are summarized in Fig. 19 for transients with overpower. The key characteristic at pressurization type transients (No.3,4) is that the power rise is very mild, because the average coolant density is not sensitive to the pressure at supercritical-pressure

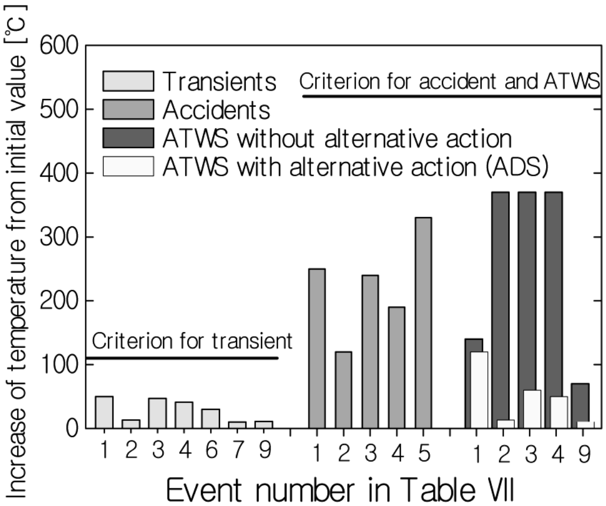

Fig. 17. Summary of Increases in Hottest Cladding Temperature

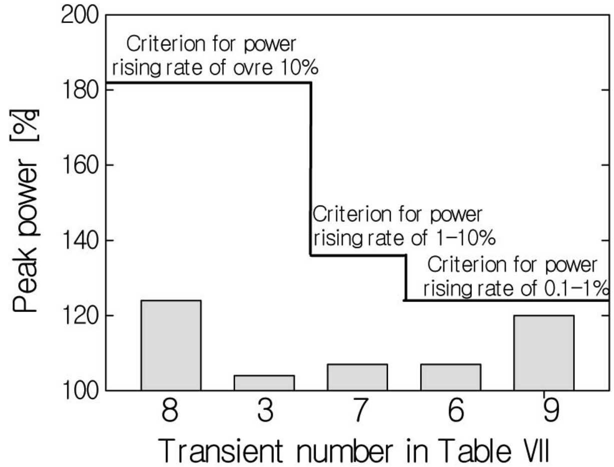

Fig. 19. Summary of Peak Powers

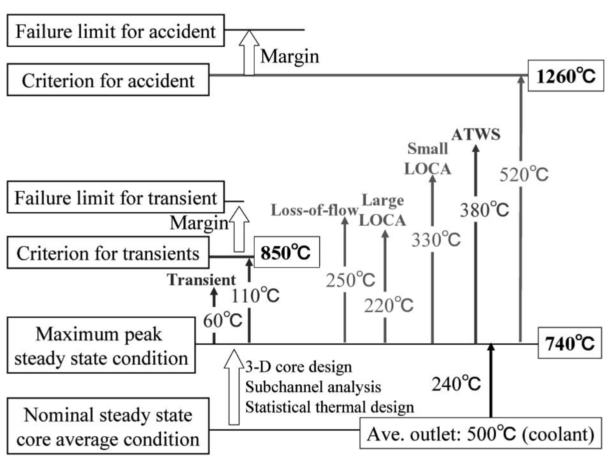

Fig.18. Increases in Maximum MCST at Loss-of-Cooling Events

where the difference in density is small between "steam" and "water", and because closing the outlet of the once-through coolant cycle causes flow stagnation in the core, which suppresses an increase in the coolant density. For overpower type accidents such as CR ejections, the criterion of the pellet enthalpy is satisfied with a satisfactory margin even though very conservative reactivity insertion is assumed. The peak pressures are summarized in Fig. 20 for events with pressure rises. The relative pressure change is small due to the high steam density and the mild power response.

The good inherent safety performance of the Super LWR is highlighted in ATWS events. Besides the above good responses against loss-of-flow and pressurization, the Super LWR has self-controllability of the reactor power against flow abnormalities and reactivity insertion due to the coolant density and Doppler feedbacks. All the ATWS events satisfy the same criteria as those of accidents with an acceptable margin and settle to a high-temperature stable condition without any alternative action, even for a high power rating core. This is an

Table 8. Duration of High Cladding Surface Temperature at Abnormal Transients

Abnormal transients	Duration of high cladding surface temp.[s]	
	> Initial temp. + 20°C	> Initial temp. + 40°C
Loss of feedwater heating	6.3	-
Partial loss of reactor coolant flow	4.9	2.5
Loss of turbine load (without bypass)	1.2	0.1
Isolation of main steam line	0.8	0.1

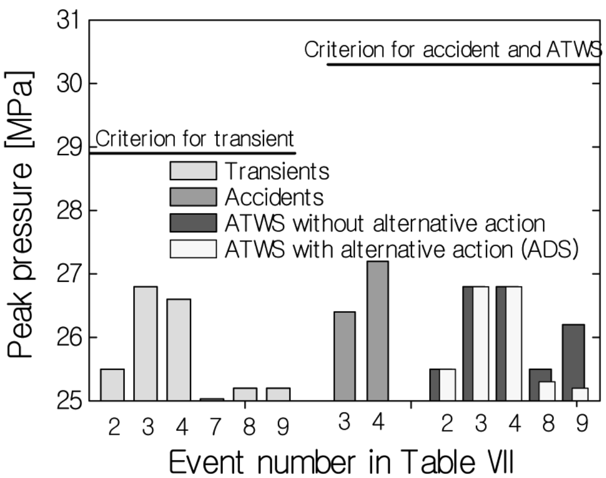

Fig. 20. Summary of Peak Pressures

outstanding safety advantage of the Super LWR. Opening the ADS valves by detecting scram failure would be an effective alternative action to increase the safety margin in ATWS [9].

The changes in the maximum cladding surface temperature (MCST) at representative loss-of-cooling events are summarized in Fig. 18 along with the change in temperature at normal operating conditions. As is shown in this figure, we comprehensively evaluated the MCST, the key parameter for SCWR design and safety, by a series of studies, i.e., core design [10], subchannel analyses [19], statistical thermal design [20], and safety analyses presented in this paper.

8. CONCLUSIONS

Safety studies on the Super LWR, a pressure-vessel type thermal spectrum SCWR, are summarized. In contrast with LWRs, the appropriate safety principle for the Super LWR is not inventory control but rather flow rate control. There are key safety characteristics of the Super LWR that inhere in the design features and they have been identified through a series of safety analyses. In the case of loss-of-flow type accidents, fuel rod heat-up is mitigated by the "heat sink" and "water source" effects of the water rods. The cladding temperature is kept lower in loss-of-flow type transients due to scenario separation from the accident in consideration of scram timing. The response of the reactor power against pressurization events is mild due to a small sensitivity of the average coolant density to the pressure and the flow stagnation of the once-through coolant cycle. The relative pressure change is also small due to the high steam density and the mild power response. The duration of the high cladding surface temperature is very short at abnormal transients. Opening the ADS valves provides effective heat removal from the fuel rod. The "in-vessel accumulator" effect of the reactor vessel top dome enhances the fuel rod cooling. A large LOCA is mitigated by the ADS. The key inherent safety characteristic is that the Super LWR does not need alternative actions to satisfy the safety criteria for ATWS events, even for a high power rating core. It is anticipated that the findings of these studies will be utilized in future R&D of the SCWR.

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