SAFETY ANALYSIS OF JSCWR

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Abstract

In order to assess viability of the Japanese Super-Critical Water-cooled Reactor (JSCWR) concept, which is a pressure-vessel type, thermal spectrum SCWR, a series of safety analyses have been conducted. The safety system of the JSCWR is designed with referring to that of BWRs. Ten abnormal transients and five design basis accidents, including LOCAs, are selected from those of current LWRs. Although the safety analyses here are not for licensing but for developing and understanding the JSCWR concept, the trend of the analysis results implies that there is no critical problem in the safety of the JSCWR concept from the viewpoint of its viability and it will encourage the next R&D step of the JSCWR.

1. Introduction

A Japanese consortium consisting of the Institute of Applied Energy, the University of Tokyo, Kyushu University, Kyoto University, Japan Atomic Energy Agency, Hitachi-GE Nuclear Energy, Ltd., Hitachi, Ltd., and Toshiba Corporation has been working together to establish the Japanese Super-Critical Water-cooled Reactor (JSCWR) concept and to implement key technology developments [1,2]. The core and plant design specifications are introduced in other papers in this symposium [3,4].

The present project on the JSCWR is the last stage of the “viability phase”. Although the R&D of the JSCWR is still conceptual and hence its detailed safety analysis for licensing is not possible, it is important to understand at the present stage whether there are critical problem(s) on safety of the JSCWR or not in order to judge its viability for going to the next R&D step. This paper describes the safety principle, safety system design, safety criteria, selection of abnormal events and safety analysis results. They are mostly based on the previous studies on the Super Light Water Reactor (Super LWR) by the Univ. Tokyo [1,5-9].

2. Safety principle for core cooling

The coolant cycle of the JSCWR is compared with those of existing power plants in Fig. 1. LWRs have a coolant circulation system, i.e., the primary system of PWRs and the recirculation system of BWRs. One of the fundamental safety requirements for LWRs is different from that for the JSCWR. That for LWRs is keeping sufficient coolant inventory in the circulation system so as to flood the core and remove heat by either forced or natural circulation. The coolant inventory is maintained by monitoring the water level in the reactor vessel of BWRs or the pressurizer of PWRs. Since the JSCWR is cooled by single-phase flow, there is no water level in the reactor vessel. Also, the JSCWR is not equipped with a pressurizer. It is difficult to monitor the coolant inventory inside the pressure boundary. However, the coolant inventory is not one of the fundamental safety requirements for the JSCWR because the once-through coolant cycle is not a closed circulation system.

The inlet and outlet of coolant in the pressure boundary are separated. The coolant enters the pressure boundary from the inlet pumps and goes out through the outlet valves. In consideration of these design features, the safety principle for core cooling for the JSCWR must be 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 [1,5,6].

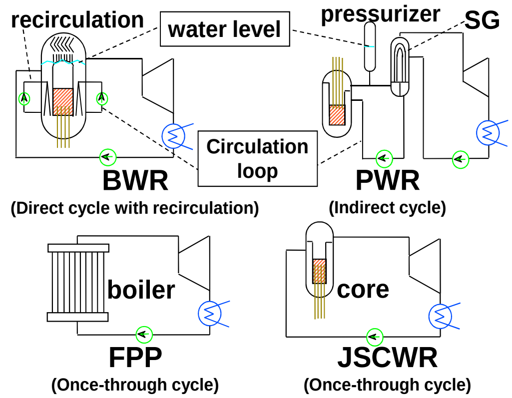


Figure 1 Comparison of coolant cycles.

3. Safety system design

The safety system of the JSCWR is schematically described in Fig. 2. The Emergency Core Cooling System (ECCS) of the JSCWR consists of the Auxiliary Feedwater System (AFS), Low Pressure Core Injection System (LPCI), and Automatic Depressurization System (ADS). A simplistic configuration of the AFS and LPCI is shown in Fig. 3. It is based on that for advanced BWRs (ABWRs). The equipment of the safety system is introduced below.

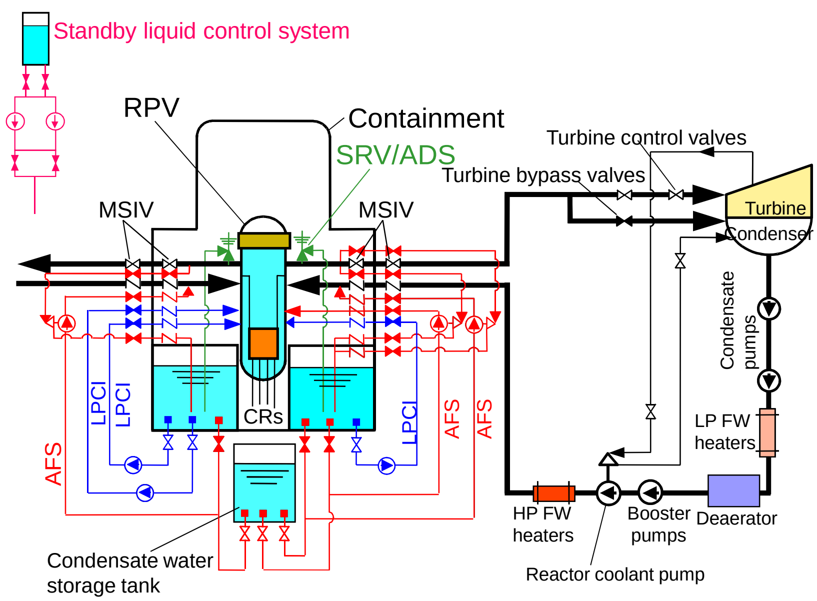
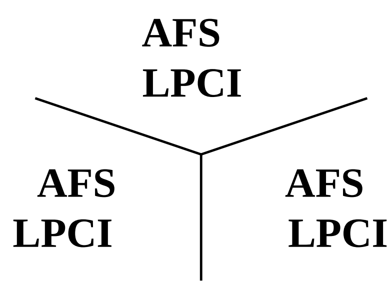


Figure 2 Plant system of JSCWR.



AFS (4% capacity x 3 units)
ADS (20% capacity x 8 units)
LPCI (12% capacity x 3 units)

100% corresponds the rated main coolant flow rate.

Figure 3 ECCS design.

3.1 Reactor shutdown system

For reactor shutdown, the reactor scram system and the Standby Liquid Control System (SLCS) are prepared in the same manner as in BWRs. The scram reactivity curve that has been calculated by a 3D neutronics calculation [4] is applied to the safety analyses. The signal delay is assumed as 0.55s which is equal to the longest one in ABWR design, excluding the signal delay for "water level low" (remember that the JSCWR has no water level in the pressure boundary). The reactivity inserted through the reactor scram is calculated by the 3D neutronics calculation [4] as about -16%dk/k. It is about three times larger than that of PWRs because the chemical shim control cannot be used in the JSCWR. However, it is smaller than that of typical BWRs. The SLCS is provided for backup shutdown.

3.2 Coolant supply system

Two turbine-driven Reactor Coolant Pumps (RCPs) are provided for normal operation. For plant startup and backup of the turbine-driven RCPs, two motor-driven RCPs with half the capacity of that of the turbine-driven RCPs are provided as in BWRs.

Three trains of the AFS are provided for the backup of these RCPs. It should be noted that the motor-driven RCPs are not credited in the safety analyses just as they are not credited in BWRs. The capacity of a single train is 4% of the rated flow; this is determined on the basis of removing the decay heat up to 6% of the rated power by two trains considering a single failure. The AFS also plays the role of Reactor Core Isolation Cooling (RCIC) because the main steam is extracted upstream from the Main Steam Isolation Valves (MSIVs). The start time of the AFS is determined by reference to the turbine-driven RCIC of ABWRs.

Three trains of the LPCI are provided for the backup of the AFS and mitigation of Loss Of Coolant Accident (LOCA). The LPCI is one of the functions of the residual heat removal (RHR) system. Three trains are provided. The capacity of the single train is 12% of the rated flow that is determined in order to keep the Peak Cladding Temperature (PCT) well below the safety criterion even with single failure. The emergency diesel generators supply electric power to the LPCI even if the offsite power is lost. In that case, 30s is assumed as the start time of the emergency diesel generators.

3.3 Valves for coolant discharge and isolation

For the discharge of coolant, safety relief valves (SRVs) are prepared in case of a turbine trip without bypass or MSIV closure. The SRVs also act as the ADS, as in BWRs.

One of the advantages of the once-through coolant cycle is that depressurization cools the core effectively. The ADS lends unique behavior to the SCWR [1,5,7-9]. Initiating the ADS induces strong core coolant flow in the core. This safety characteristic derives from the once-through coolant cycle. After depressurization, the core is cooled by the LPCI. Although maintenance of the supply of coolant from the cold-leg and discharge of coolant at the hot-leg are required for decay heat removal on a long time scale, the core can be cooled by reactor depressurization for short times up to 1-2 minutes according to the reactor vessel size without the supply of coolant from the cold-leg. This allows the emergency diesel generators for the LPCI or the RHR to have a realistic start time similar to the 30s in LWRs.

The function of the MSIVs in the JSCWR is to avoid release of radioactivity outside of the containment as in BWRs. The MSIV characteristic is determined to be the same as that of ABWRs where the valves close in 3s from the signal including the delay of actuation.

4. Actuation conditions of the safety system

The actuation conditions of the safety system are summarized in Table 1. Abnormalities in supplying coolant from the cold-leg are mainly 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 rates, the reactor is depressurized, and then cooled by the LPCI.

In case of a large break of the cold-leg pipe, it is necessary to detect leakage of the coolant and to open the ADS valves in order to recover the coolant flow in the core. Although detecting a decrease in the core pressure would work well in case of LOCA, this signal also leads to opening the ADS valves at the pressure-decreasing transients such as abnormal opening of the turbine control valves. Fast depressurization at abnormal transients may have problem(s) for reuse of the reactor vessel. Therefore, only reactor scram is actuated by detecting a decrease in the core pressure. Although an increase in the pressure or radiation level in the containment should be detected in case of LOCA, it is not credited in the safety analysis here. When the measured main steam flow rate is smaller than the main coolant (feedwater) flow rate, it is suspected that coolant is leaking. The reactor is tripped when the mismatch of these flow rates reaches 10% of the measured main coolant flow rate. The reactor is depressurized when the mismatch reaches 20%.

By applying the conditions in Table 1, the ADS valves are not opened in case of the pressure-decreasing transients and hence the core pressure slowly passes the critical point. It is known that the critical heat flux and minimum heat transfer coefficient is especially small just below the critical pressure. However, the safety analysis result (see section 6.4.4) shows that the fuel rods are not heated-up.

Table 1 Summary of actuation conditions of safety system.

	Actuation conditions				
Reactor scram	**Pressure high (level 1)**	Reactor power high			
	Pressure low (level 1)	**Main coolant flow rate low (level 1)**			
	MSIV closure	**Main coolant / main steam flow mismatch (level 1)**			
	Main stop valve closure	Turbine control valve quickly closed			
	ECCS start-up	Reactor coolant pump trip			
	Drywell pressure high	Earthquake acceleration large			
	Drywell radioactivity high	Loss of offsite power			
	Reactor period short	Condensate pump trip			
AFS	Loss of offsite power	**Main coolant flow rate low (level 2)**			
	Condensate pump trip	**Main coolant / main steam flow mismatch (level 1)**			
	MSIV closure	Turbine control valves quickly closed			
	Main stop valves closure	Reactor coolant pump trip			
SRV	Relief valve function			Safety valve function	
	Open (MPa)	Close (MPa)	Number	Open (MPa)	Number
	26.2 (Pressure high level 2)	25.2	1	27.0	2
	26.4	25.4	1	27.2	3
	26.6	25.6	3	27.4	3
	26.8	25.8	3		
ADS, MSIV, LPCI	**Main coolant flow rate low (level 3)**			Drywell pressure high	
	Main coolant / main steam flow mismatch (level 2)			Drywell radioactivity high	

5. Safety criteria

Since the JSCWR is presently in the concept development phase, the safety criteria cannot be determined based on experiments. The principle for the safety criteria and tentative values for the safety analyses have been determined. The requirements for abnormal transients (Conditions II and III in the international category) are the same as those of LWRs: no fuel rod damage and no pressure boundary damage. The requirements for accidents (Condition IV) are no excessive core damage and no pressure boundary damage, which also correspond with LWRs. The safety criteria described below are determined for concept development.

5.1 Criteria for fuel rod integrity

As described in the Reference [10], the criteria for mechanical integrity of the fuel rod cladding during normal operation and abnormal transients are:
(1) Thickness of cladding tube shall be sufficient to avoid buckling collapse.
(2) Von Mises equivalent stress in cladding shall be lower than the yield strength (S_y) of the cladding material in normal operation and abnormal transient.
(3) Cumulative creep Damage Fraction (CDF) based on the cladding material shall be smaller than 0.3 in normal operation and abnormal transient.
(4) Pellet center line temperature shall be lower than the melting point of fuel with effect of burnup in normal operation and abnormal transient.
(5) In the case of over inner pressure by FP gas release, out-going creep displacement shall be prevented so that pellet-cladding gap can keep its integrity from thermal feedback.

The cladding thickness is determined in order to satisfy the criterion (1). The criteria (2) to (5) are checked by the fuel rod thermal/mechanical analyses [10] where the safety analysis results here (time sequences of the cladding surface temperatures, the relative power and the coolant pressure) are used as the boundary conditions.

Since heat transfer deterioration is a much milder phenomenon than boiling transition, it is not necessary to take the deterioration heat flux ratio for avoiding the cladding thermal failure, unlike the critical heat flux ratio of LWRs. It is better to directly limit the maximum cladding surface temperature (MCST) for avoiding phase transformation of austenitic stainless steel. In the present analysis, the criterion of MCST is set as 800°C, which is well below its phase transformation temperature and also lower than the criterion in typical fast breeder reactors.

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 early US PWRs with stainless steel cladding.

A reactivity insertion over $1 is not expected in the JSCWR, because the reactor is tripped by detecting high neutron flux level or short reactor period before a CR blade is fully withdrawn or drops. Thus, the maximum allowable temperatures of the cladding instead of fuel enthalpies are taken as the criteria for the abnormal transients and accidents, respectively.

5.2 Criteria for pressure boundary integrity

The relative pressure change in SCWRs is smaller than that in LWRs due to the once-through coolant cycle and the high operating pressure. The maximum allowable pressures for abnormal transients and accidents are set at 105% and 110% of the design pressure (27.5MPa), respectively, while those of LWRs are 110% and 120%.

6. Safety analyses

6.1 Selection of abnormal events

Since the JSCWR is also a light water cooled reactor and the components are similar to those of LWRs, its abnormalities are taken from those of LWRs. The abnormal events of the JSCWR are summarized in Table 2. All the events are taken from either PWRs or BWRs [1,6,9].

6.2 Safety analysis codes

Three different codes that were developed at the Univ. Tokyo [1,6-9], are modified and applied to the JSCWR geometry. They are the system analysis code for supercritical-pressure conditions (SPRAT-DOWN), the system analysis code for depressurization conditions (SPRAT-DOWN-DP) and the reflood analysis code (SCRELA reflood module). The calculation model of the system analysis code for supercritical-pressure conditions is schematically shown in Fig. 4.

Table 2 Analyzed events.

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

*Results of **bolded** events shown below

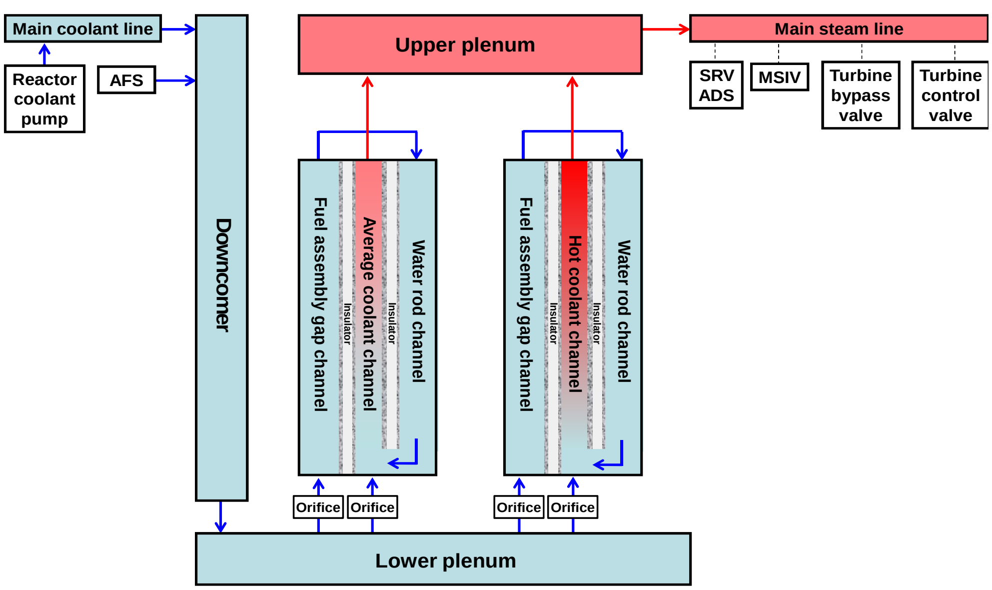


Figure 4 Calculation model of system analysis code for supercritical-pressure conditions.

6.3 Safety analysis conditions

Table 3 shows the initial conditions of the average and hot channels for the safety analysis. The initial MCST of 697°C is equal to the core design result [4] in consideration to engineering uncertainty with 99.99% probability and 95% confidence level. Since the reactivity feedback coefficients (coolant density and Doppler) does not significantly change with the burnup, there is very small difference in the reactor behavior between the Begin-Of-Cycle (BOC) and End-Of-Cycle (EOC), only the results at BOC are introduced in this report.

Operation of plant control system is considered in the safety analysis as in BWRs since the JSCWR also adopts a direct steam cycle while PWRs neglect it. The main steam pressure, the main steam temperature and the reactor power are regulated by the turbine control valves, the reactor coolant pumps and the control rods, respectively [1,11].

Table 3 Initial conditions of average and hot channels.

	Average channel	Hot channel
Average linear heat generation rate [kW/m]	13.5	27.0
Maximum linear heat generation rate [kW/m]	-	38.5
Mass flux at coolant channel [kg/m^{2}s]	1100	1408
Mass flux at fuel assembly gap channel [kg/m^{2}s]	456	584
Mass flux at water rod channel [kg/m^{2}s]	745	954
Feedwater temperature [°C]	290	-
Fuel assembly gap channel outlet temperature [°C]	325	332
Water rod channel outlet temperature [°C]	344	344
Coolant channel inlet /outlet temperature [°C]	310 / 510	314 / 625
Maximum cladding surface temperature [°C]	-	697

6.4 Abnormal transient analyses

6.4.1 Uncontrolled CR withdrawal at normal operation

The reactivity worth of the withdrawn CR is assumed as $1.3 which is given from the 3D core design [3]. The withdrawal speed of 3.0cm/sec is assumed. The CR is withdrawn until the reactor power reaches the scram setpoint (115% of rated power) which would be detected by neutron flux monitors. It is conservatively assumed that the CR neighboring to the hot channel is withdrawn and that a signal of an increase in the local neutron flux fails. The increase in the power peaking factor of the hot channel is calculated by 3D steady-state calculations [4] in advance and conservatively applied to the safety analysis. It means that mitigation of the power peaking by reactivity feedback is neglected. When the reactor power reaches the peak value, the peaking factor of the hot channel increases by about 4% from the initial value in the present analysis. The calculation result is shown in Fig. 5. The MCST is 793^{o}C and the maximum reactor pressure is 25.2 [MPa].

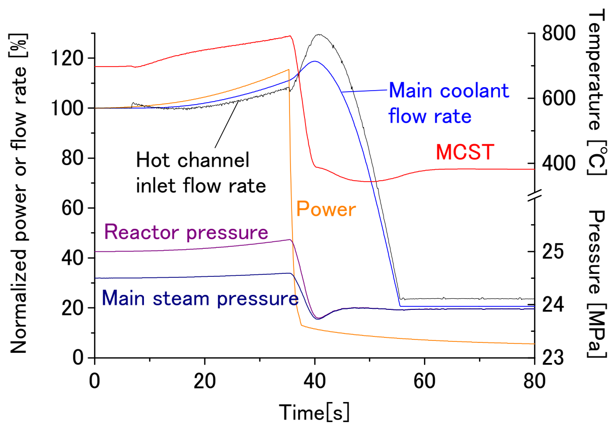


Figure 5 CR withdrawal at normal operation

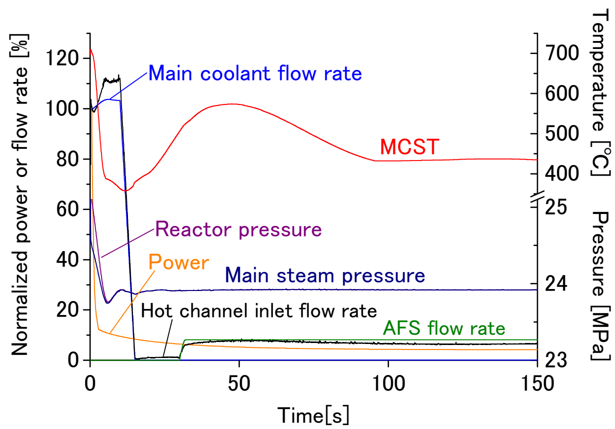


Figure 6 Loss of offsite power.

6.4.2 Loss of offsite power

This is the typical transient with trip of both RCPs. The motor-driven condensate pumps are assumed to trip instantaneously. The turbine control valves are quickly closed due to a turbine trip. The turbine bypass valves open immediately after that. A scram signal and an AFS signal are released by detecting the “loss of offsite power” or “turbine control valves quickly closed” or “condensate pump trip.” Both RCPs are assumed to trip at 10s due to a decrease of the water level in the deaerator or loss of steam to the turbine-driven RCPs. By considering single failure, two of three AFS units are initiated at 30s. The calculation results are shown in Fig. 6.

At the beginning, the cladding temperature and the pressure temporarily increase due to the closure of the turbine control valves. Then, they decrease due to the turbine bypass. After the trip of the RCPs, the cladding temperature increases again. After two-out-of-three AFS units start up, the cladding temperature decreases again. The MCST is 710°C. The maximum pressure is 25.3 [MPa]. Both peaks appear at very beginning of the event.

6.4.3 Loss of turbine load

When the turbine bypass is credited, the analysis scenario is the same as that of the ‘‘loss of offsite power.” Only the case without the turbine bypass is analyzed. This event is a typical pressurization transient. The reactor behavior is shown in Fig. 7. A scram signal and an AFS signal are released by detecting the “turbine control valves quickly closed”. Since the turbine bypass fails, the pressure quickly increases. The peak pressure is only 107% of the initial value. Unlike BWRs, the reactor power does not increase before the reactor scram. One reason is that the density difference between supercritical “water” and “steam” is much smaller than that between saturated water and steam at the BWR operating pressure. The other is that flow stagnation in the core due to the closure of the turbine control valves causes an increase in the coolant temperature which avoids the increase in the coolant density and the power. When opening the SRVs, the pressure and power begin to decrease. The MCST is 761°C. The maximum reactor pressure is 26.7 [MPa]. After the pump trip at 10s, the reactor behavior is similar to that at “loss of offsite power”. Single failure is assumed for the AFS.

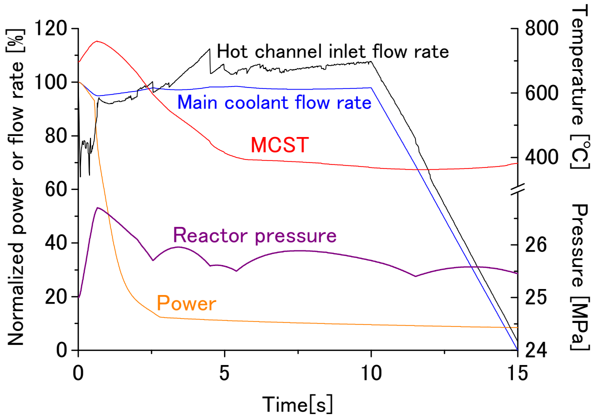


Figure 7 Loss of turbine load

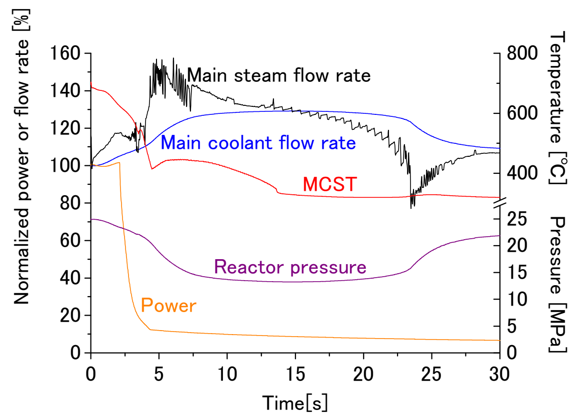


Figure 8 Pressure control system failure

6.4.4 <u>Pressure control system failure</u>

This is a typical pressure decreasing transient. The maximum opening signal (130% of rated value) is assumed to given to the turbine control valves. The result is shown in Fig. 8. The reactor is scrammed when the pressure reaches the low level 1 (24.0MPa). Although the pressure decreases to subcritical region and hence two-phase flow appears in the core, the MCST does not increase due to the reduced heat flux by the scram.

6.5 Design basis accident analyses at supercritical pressure

6.5.1 <u>Total loss of reactor coolant flow</u>

This accident is defined as a simultaneous trip of both RCPs. The main coolant flow rate decreases linearly to zero in 5s. The scram signal is released by detecting ‘‘flow rate low level 1’’ at 0.5s. Although the trip of the RCPs itself would release the scram signal, it is conservatively neglected. The AFS signal is released at 0s and the actuation of the AFS with single failure is assumed to start at 30s. The calculation result is shown in Fig. 9. The MCST is 974°C and is well below 1260°C.

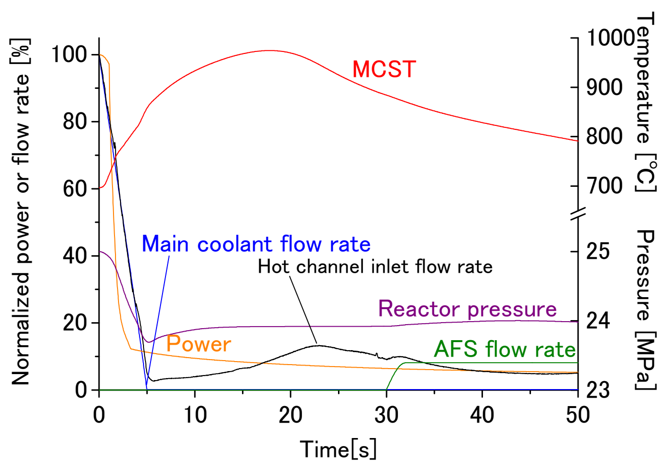


Figure 9 Total loss of reactor coolant flow.

6.6 Loss of coolant accident analyses

6.6.1 <u>Large LOCA</u>

The large LOCA is defined as a pipe break followed by an increase in the mismatch between the main coolant and main steam flow rates to the Level 2 (20%). When the break area of the cold-leg pipe is larger than 3% of its cross section, it is a large LOCA in the present design. The coolant flow during blowdown is described in Fig. 10. Before the ADS valves are opened, the cladding temperature increases because flow stagnation occurs at the upper part of the core.

After the ADS valves are opened, the core coolant flow recovers and the cladding temperature decreases. Two of three units of the LPCI, by considering single failure, are actuated when the pressure decreases below 0.8MPa. When the pressure reaches the containment pressure or the coolant from the LPCIs fills the bottom plenum (refill completed), the blowdown calculation is finished. The calculation results of the blowdown phase with 100% and 5% breaks are shown in Fig. 11 as the examples. As described above, the cladding temperature quickly increases before opening the ADS valves and then decreases after opening them. The MCST does not exceed 1000°C for all the cases. When the pressure is near the containment pressure and hence the flow rate is almost zero in the core, the cladding temperature increases again.

As the break area is larger, the reactor depressurization is faster and hence the core must wait for the water from the LPCIs for longer time. Thus, the 100% break gives the highest MCST in the reflood phase. The reflood phase is schematically described in Fig. 12. The calculation result of the reflood phase with 100% break is shown in Fig. 13. The core is quenched in about 1500s. The MCST is about 1050°C that is still well below the criterion of 1260°C.

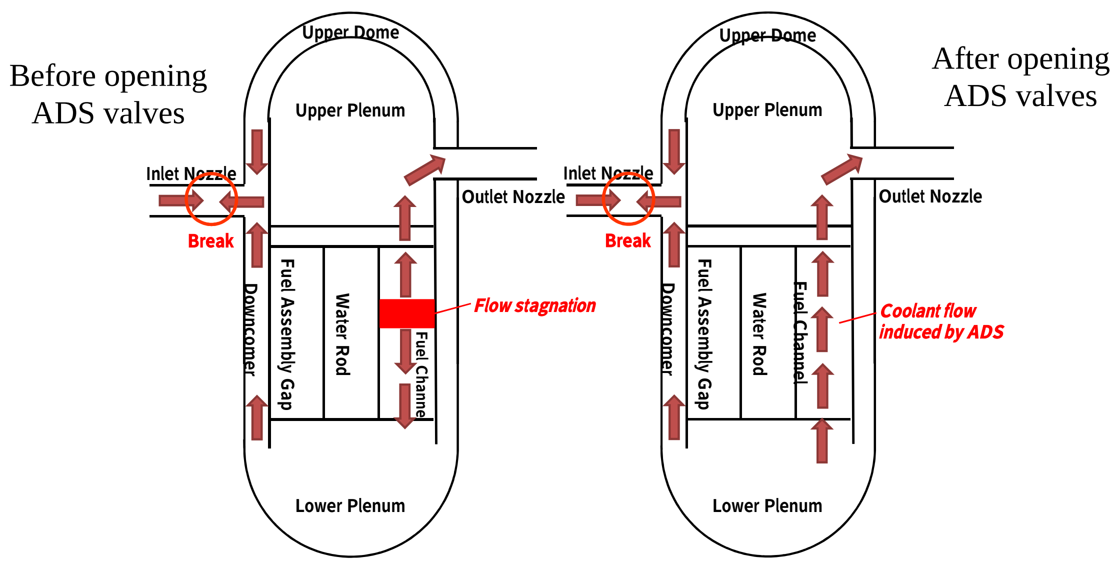


Figure 10 Coolant flow during blowdown at cold-leg large break LOCA

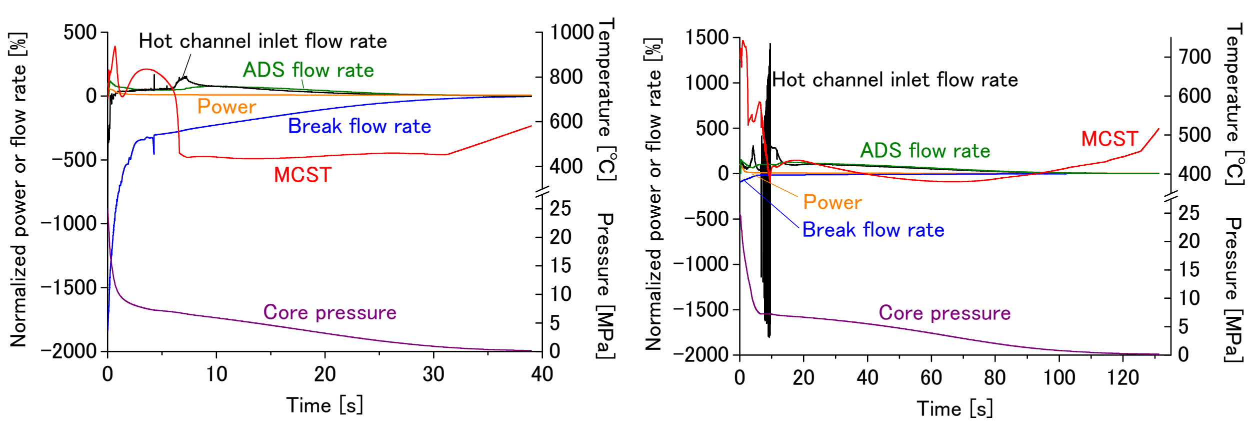


Figure 11 Blowdown phase of cold-leg large break LOCAs. (Left: 100% break, Right: 5% break)

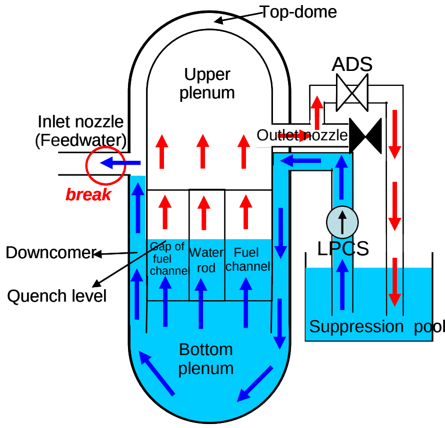


Figure 12 Coolant flow during reflood

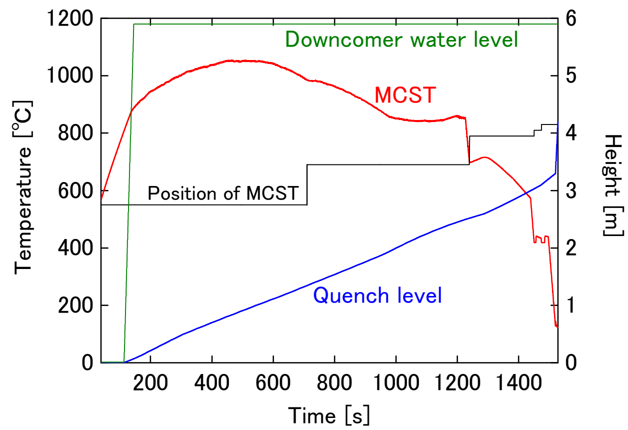


Figure 13 Reflood phase of 100% LOCA.

6.6.2 Small LOCA

The small LOCA includes the pipe breaks where the break area is smaller than that defined as the large LOCA. The calculation results are shown in Fig. 14. The ADS valves are not opened as the "Level 2" setpoint of the inlet/outlet flow mismatch is not detected. The core heat-up is mild since the leak flow rate is small. The highest MCST is 854°C in the case of 2% break.

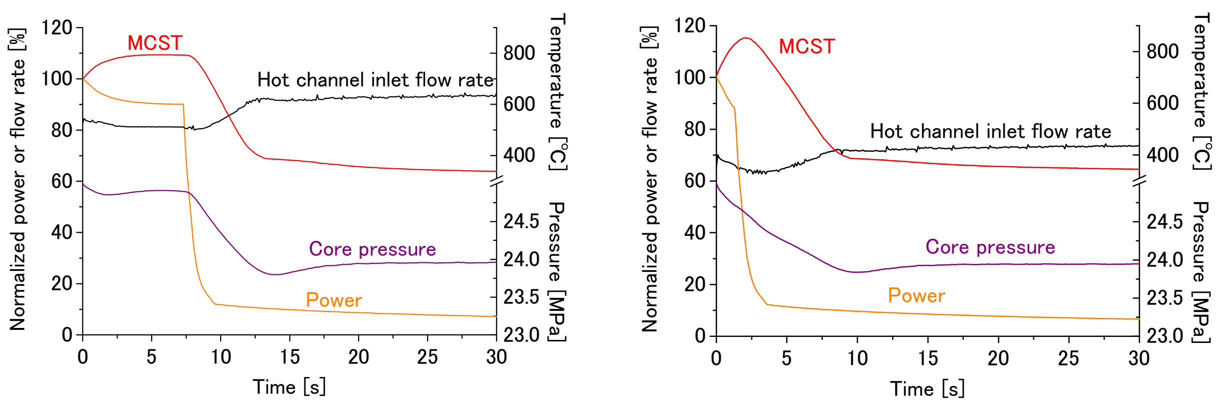


Figure 14 Cold-leg small break LOCAs. (Left: 1% break, Right: 2% break)

6.7 Summary of safety analyses

The safety analyses have shown that the criteria of the MCST and pressure are both satisfied for all the abnormal transients and design basis accidents. The MSCTs calculated through the safety analyses are summarized in Fig. 15. The maximum increase in the MCST from the normal operating condition is about 100 °C and 350 °C in the abnormal transients and design basis accidents, respectively. For the abnormal transients, the mechanical integrity of the fuel cladding is assessed by the fuel rod thermal/mechanical analyses [10].

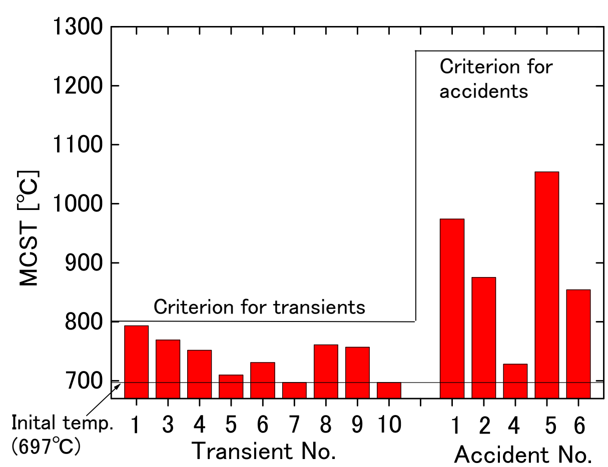


Figure 15 Summary of MCSTs
(Event numbers correspond to Table 2.).

7. Conclusion

A series of safety analyses were carried out based on the core design and safety system design of the JSCWR. The MCST did not exceed the criteria even though the initial value was based on the subchannel analysis result with 99.99% probability and 95% confidence level. The pressure increases by only 7% in the limiting event. The fuel rod mechanical integrity was also assessed by the fuel rod analysis in the Reference [10]. It is concluded that no critical problem has been found for the safety of the JSCWR from the viewpoint of its validity and it will encourages the next R&D step of the JSCWR including more detailed safety analyses and related sensitivity analyses.

Acknowledgement

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