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Preliminary Safety Evaluation of Korean 600MWe Liquid Metal-cooled Reactor to ATWS Events

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Abstract

The inherent safety of a Korean liquid metal reactor having a net electric rating of 600 MW is evaluated against some anticipated transients without scram (ATWS). The main goal of this design is the satisfaction of the enhanced economics and the proliferation-resistance. One of the remarkable enhancements of the design is the unique core layout with no blanket region. The peak fuel temperatures for the three typical ATWS events of UTOP, ULOF, and ULOF/LOHS are well within the temperature limit of fuel. The clad and sodium temperatures are also do not exceed the safety criteria during the transients. Therefore, the inherent safety characteristics of this design are ensured by the present analyses. Further, it is estimated that the evolutionary core design and other design efforts well accommodate the increased reactor power to guarantee the safety of the Korean liquid metal reactor of 600 MW power.

1. Introduction

Recently international efforts are concentrated to develop nuclear systems satisfying the goal of economics and proliferation-resistance. Reflecting these efforts, a Korean liquid metal reactor with increased electric power of 600 MW is under development. To accommodate the increased power of 600 MWe the heat transport systems are changed considerably compared to those of KALIMER-150 [1]. The intermediate heat transport system (IHTS) of the preliminary design consists of 3 loops, thus there are 3 steam generators (SGs) in the design. The passive decay heat removal circuit (PDRC) is additionally equipped to remove the decay heat when the sodium level in the hot pool increases. Therefore, the reactor is cooled down by the combined function of PDRC and the passive vessel cooling system (PVCS), which constitute the passive safety decay heat removal system (PSDRS), when there occurs an off-normal or accidental situation. One of the important design targets of the reactor is the proliferation-resistant characteristic. For this, the core is designed not to have any blanket but still to maintain the critical breeding ratio [2]. This design eliminates the possibility of excess Pu material production. The breakeven core design without blanket can be achieved due to the good internal breeding characteristics of metalic fuel. Figure 1 depicts the layout of the breakeven core of the reactor.

The main purpose of the present analysis is the validation of core design parameters, fluidic design and mechanical design parameters of the new design. Another important purpose of the analysis is to show the feasibility of the newly introduced system such as the passive decay heat removal circuit (PDRC) as a part of the total plant system. The final purpose is the evaluation of the inherent safety of the design against some anticipated transients without scram (ATWS). Three major ATWS scenario, the unprotected transient over-power (UTOP), the unprotected loss of flow (ULOF) and the unprotected loss-of-flow with loss-of-heat sink (ULOF/LOHS), are analyzed.

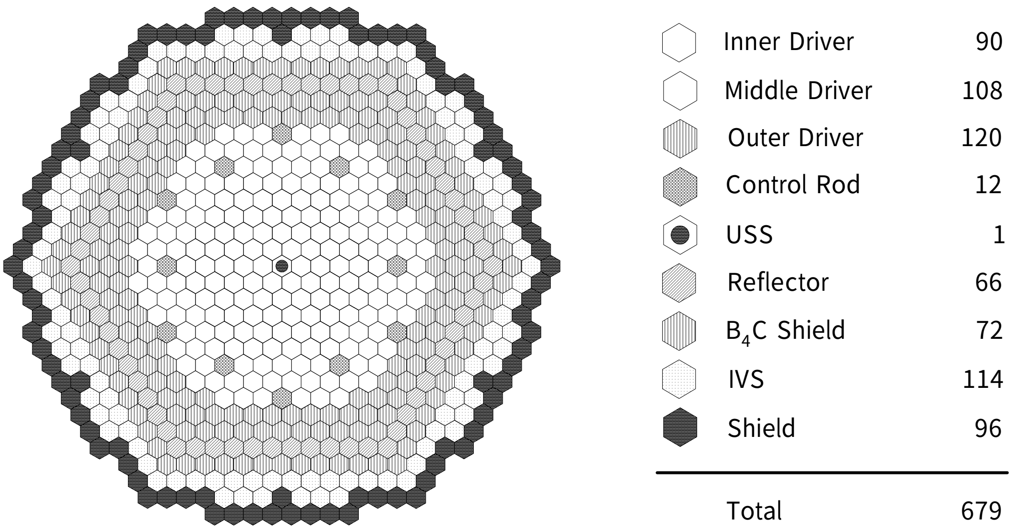

Figure 1. Breakeven Core Layout of the preliminary design

2. Analysis Method

The SSC-K code is used for assessment of the inherent safety the Korean liquid metal cooled reactor of 600 MW power. The SSC-K aims at not only extensive analysis capability and flexibility, but also efficiently fast running enough to simulate long transients in a reasonable amount of computer time. The code thus becomes capable of handling a wide range of transients, including normal operational transients, shutdown heat removal transients, and hypothetical ATWS events. The SSC-K code [3] has been developed by KAERI for the analysis of system behavior during transients. The SSC-K code features a multiple-channel core representation coupled with a point kinetics model with reactivity feedback. It provides a detailed, one-dimensional thermal-hydraulic simulation of the primary and secondary sodium coolant circuits, as well as the balance-of-plant steam/water circuit. Some model for PDRC is modified for the analysis of the new design with increased power of 600 MW. The code is currently being used as the main tool for system transient analysis in the liquid metal reactor development in Korea.

2.1 Assumptions

It is assumed that the unprotected transients are initiated at the full reactor power with equilibrium decay heat levels. The reactor protection system and the reactor control system are not credited unless the function of those systems results in conservative consequences. For the analysis of the UTOP event it is assumed that the reactivity of 30 cents is inserted by the removal of the control rods under the operation of the control rod stop system (CRSS). The value of 30 cents was also used for KALIMER-150. This is conservative enough considering the reduced reactivity worth of control rods in the design.

The transient of ULOF is initiated by all primary pump trips followed by coastdown. The RPS is assumed to fail to detect the mismatch between the high flux and the low flow rate and to insert the control rods. It is noted that the gas expansion models (GEMs) are not equipped in the design. In the analysis of the combined ULOF/LOHS, the transient is also assumed to initiate at full power and by the trips of all primary pumps. Further, the flow through the intermediate heat transport system (IHTS) stops and the heat generated in the core is not removed through SGs.

2.2 Analysis inputs

The important core design data are the reactivity worth described in Table 1. They describe the Doppler reactivity change, the radial expansion effect, the sodium void effect and the control rod reactivity change due to the rod insertion and/or axial expansion. The reactivity data for the BOEC are used in the steady state and transient calculation. Some components of reactivity worth are larger at BOEC than those at EOEC. Other components are more effective at EOEC. In addition, it is true that the reactivity data for BOEC result in more dominant reactivity effect than that obtained with the data for EOEC for some transient scenario. And *vice versa* for other scenario. The detailed individual effects are not evaluated for the life of core. The BOEC is selected as a nominal point for the safety analysis.

The operating conditions of primary and secondary heat transport system are summarized in Table 2. The heat added by six electromagnetic pumps (EMPs) are evaluated to be about 10.6 MW. The design parameters for IHX and SG are given in Table 3 and 4, respectively. Other fluidic system design data are found in Ref.[4].

Table 1 Summary of Reactivity Worth

	BOEC	**EOEC**
Fuel Temperature(Doppler) Coefficient (d rho/ dT)		
Sodium Flooded	$-0.08835T^{-1.4}$	$-0.08567\ T^{-1.4}$
Uniform Raidal Expansion Coefficient		
(dk/k) /(R/dR)(pcm/%)	-123	-139
dk/dT (x 10^{-4})(1/K)	-7.1520	-7.1998
Sodium Void Effect (pcm)		
Inner Driver Fuel (IDF)	1392.24	1686.72
Middle Driver Fuel (MDF)	1357.90	1295.02
Outer Driver Fuel (ODF)	220.67	192.86
IDF + MDF	2734.26	2934.10
IDF + MDF + ODF	2949.00	3120.62
Control Rods (pcm)		
Inner Rod	2789.50	2691.18
Outer Rod	2351.18	2043.61
Inner+Outer Rod	5212.40	4700.92
USS (pcm)	572.10	854.13
Total Beta-effective	0.00366	0.00362

Table 2 Primary Heat Transport System Operating Condition

Design Parameter	Design Value
Reactor Power	1500 MWt
Primary sodium inlet/outlet temp	530.0℃ / 386.2℃
Primary sodium flow rate	8141.4 kg/s
Secondary sodium inlet/outlet temp	339.7℃ / 511.0℃
Secondary sodium flow rate	6853.8 Kg/s

Table 3 Intermediate Heat Exchanger Design Parameters

Design Parameter	Design Value
No. IHX	6
Type	TEMA type S
Power	256.73 MWt
1^{st} sodium flow rate	1356.90 Kg/s
2^{nd} sodium flow rate	1142.32 Kg/s
No. tubes	4320
Tube side coolant	2^{nd} Na
Shell side coolant	1^{st} Na
Tube Pitch/Diameter (P/D)	1.72
Tube length	6 m
Tube OD	12.7 mm
Tube ID	11.1 mm
Thickness of tube	0.8 mm
Configuration of Tubes	Triangular
Tube material	SS304
Diameter Flow Hole	10 mm
No. Flow Hole	8640
IHX Shell ID	1.64 m
IHX Shell Total Length	7.5 m
Pressure drop from 1^{st} inlet nozzle to outlet nozzle	24.76 kPa
Pressure drop from 2^{nd} inlet pipe to outlet pipe	54.86 kPa

Table 4 Steam Generation Parameter Summary

Steam Generator	Thermal Capacity	500 MWt(/SG)
	Steam Cycle	Once-through superheat cycle
	Sodium Inlet Temperature	511 °C
	Sodium Outlet Temperature	339 °C
	Sodium Flow Rate	2273.2 kg/s
	Steam Temperature	483.2 °C
	Steam Pressure	15.5 MPa
	Steam Flow Rate	221.137 kg/s(/SG)
	Feedwater temperature	230 °C
Tube & Bundle	Inner Diameter	16 mm
	Wall Thickness	3.5mm
	Length	58.9 m
	No. of Tubes	560
	Surface Area	2384 m2
	Pitch	50mm x 35mm (Tra. x Long.)
	Bundle Height	6.5 m

3. Results and Discussion

The evaluation of inherent safety against the unprotected accidents is the most important part of safety analysis of the new design. Because of their very low probability of occurrence and the defense in depth enough to mitigate the consequences these events are classified as bounding events (BEs). The unprotected transient over-power (UTOP), unprotected loss of flow (ULOF) with or without loss of heat sink (ULOF/LOHS) events are analyzed. The effect of several reactivity feedback mechanisms due to Doppler, sodium void, axial fuel rod expansion, radial core expansion, and axial control rod drive line expansion are combined to achieve the negative reactivity for various scenario of the unprotected events.

3.1 Unprotected Transient Overpower (UTOP)

The UTOP event refers to an off-normal condition in which reactivity insertion is accompanied due to the malfunction in the reactivity controller with the failure of reactor protection system (RPS). The best defense against the UTOP vulnerability is to minimize the reactivity insertion in case of the malfunction of the controller. For the analysis of the UTOP event it is assumed that the reactivity of 30 cents is inserted by the removal of the control rods under the operation of the control rod stop system (CRSS). The value of 30 cents was also used for KALIMER-150. This is conservative enough considering the reduced reactivity worth of control rods in the new design of increased power.

The predicted normalized power and flow after the initiation of the UTOP event are given in Fig. 2. The power reaches peak value of 1.251 times the rated power at about 30 seconds after the initiation of the transient and it is stabilized to 1.02 times the rated power. The flow is maintained almost constant.

The change of each reactivity components during the UTOP is shown in Fig. 3. The net reactivity, which is positive up to 80 seconds after the initiation of the event, turns down and it is maintained to slightly negative by the effect of negative reactivity mechanisms. Fig. 4 shows the fuel temperature at the sixth axial node from the bottom of the core in the hot assembly. The peak fuel temperature shown in Fig. 5 reaches 1113 K at 35 seconds, which is well below the fuel melting point of 1343 K (1070 ^{o}C). The peak temperature of clad is 858 K at 50 seconds, which also provide a large margin to the temperature of eutectic formation, 1063 K (790 ^{o}C). This results guarantee the substantial margin of the design for the UTOP event. Other parameters showing the system behavior during the UTOP transient such as pool temperatures and pool levels are shown in Figs. 6 and 7.

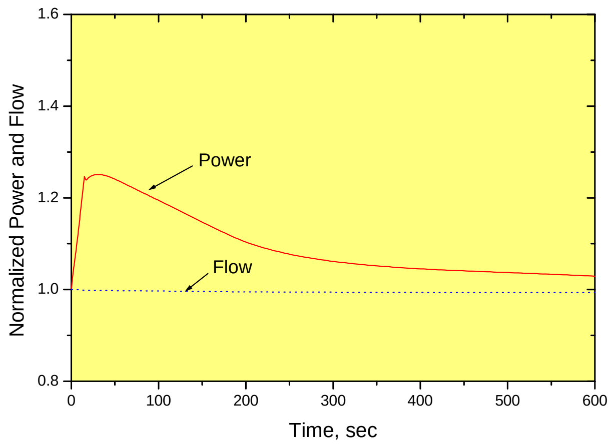

Fig. 2 Power and flow during the UTOP event

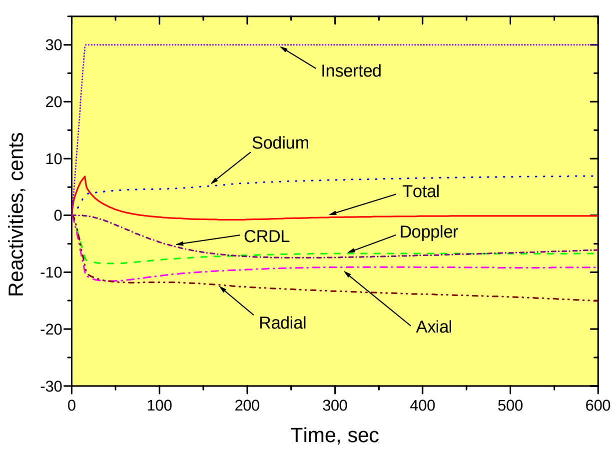

Fig. 3 The trend of each reactivity components during the UTOP event

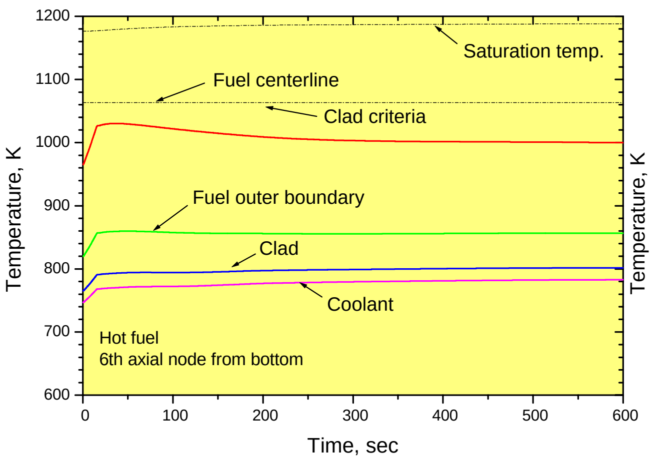

Fig. 4 Fuel temperature during the UTOP event

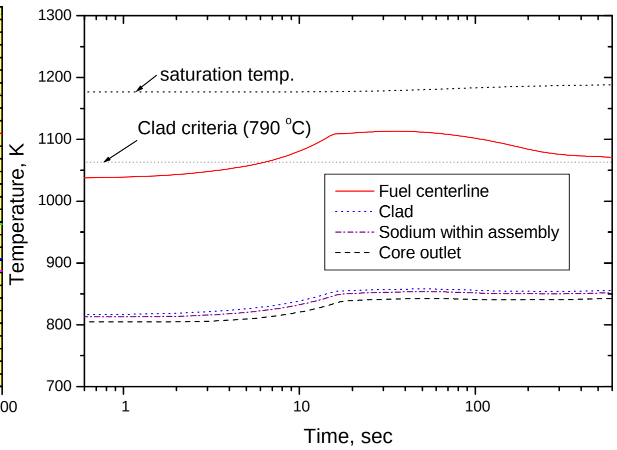

Fig. 5 Peak temperatures of fuel and coolant during the UTOP event

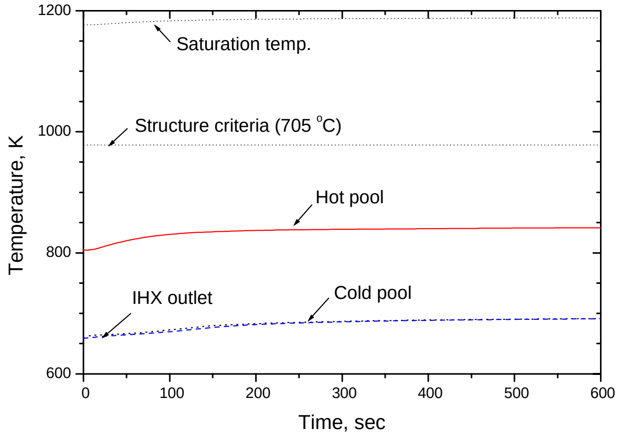

Fig. 6 Temperature of hot and cold pools during the UTOP event

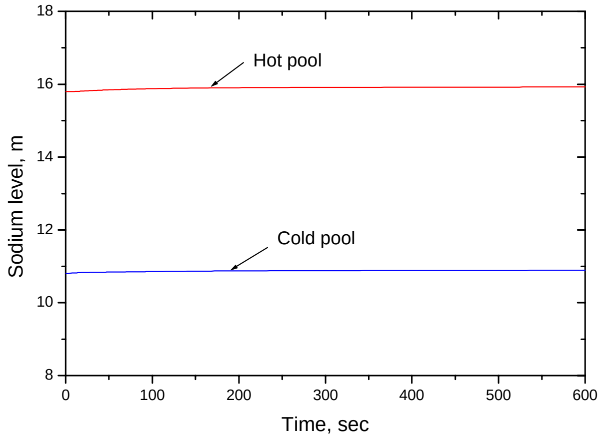

Fig. 7 Hot and cold pool levels during the UTOP event

3.2 Unprotected Loss of Flow (ULOF)

The unprotected loss of flow (ULOF) accident is initiated by the trip of all primary pumps followed by coastdown. All the primary EM pump are supported by an flow inertia device so that the PHTS has a flow coastdown characteristics to prevent the reactor core damage following a LOOP to the primary coolant pump without use of external AC power. Upon loss of the normal power supply, the stored kinetic energy in the flywheel of the flow inertia device is utilized to generate the required electricity for coastdown of the primary pump. Each inertia device is connected to each pump one by one.

The ULOF transient is assumed to occur at full power. It is also assumed that the scram signal from high flux-to-flow ratio is not generated because of the failure of the detection of the mismatch between the flux and the flow. The IHXs are normally operating so that all heat generated in the core is removed through the IHTS and SGs. The safety grade system of the PSDRS is credited throughout entire transient in the analysis. In the analysis of ULOF for KALIMER-150, the role of gas expansion module (GEM) was very effective to mitigate the consequence of the transient. However, the new design does not adopt the GEM system. Therefore, the negative reactivity feedback effects become more important in the analysis of ULOF for the 600 MW design.

The power-to-flow ratio is the key parameter that determines the consequence of a ULOF event. Fig. 8 shows the trend of normalized power and flow during ULOF event. As the core flow rate decreases the core is heated up abruptly, thus, the Doppler, axial and radial expansion components of reactivity initially turn negative. The flow reaches equilibrium near about 9 % of the initial core flow at 110 seconds and it maintains nearly constant. On the other hand the power drops to 13.1 % of the full power at 280 seconds and it increases very slowly after that. Even though the power level is higher than the flow level, the heat is removed through the normal heat removal paths. The slow increase of power after 360 second results from the small amount of net positive reactivity shown in Fig. 9. The sodium pool temperatures are given in Fig. 10. The fuel temperatures show slightly different trends depending on the axial location. The temperatures for the sixth and tenth axial nodes are given in Figs. 11 and 12, respectively. The peak temperatures in hot assembly are described in Fig. 13. The response for ULOF event is generally much different from that of KALIMER-150 excluding the effect of GEMs. Both designs satisfy the safety criteria for ULOF.

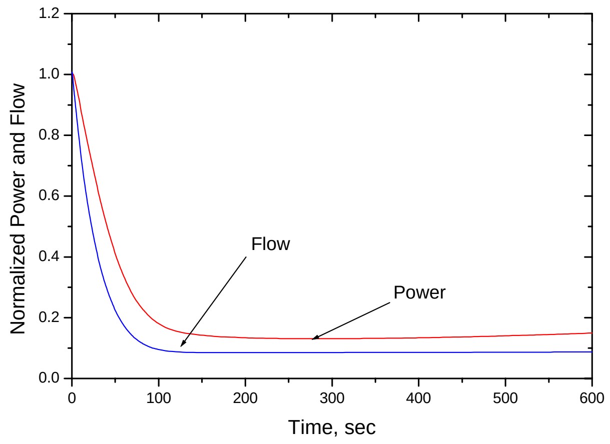

Fig. 8 Normalized power and flow during the ULOF event

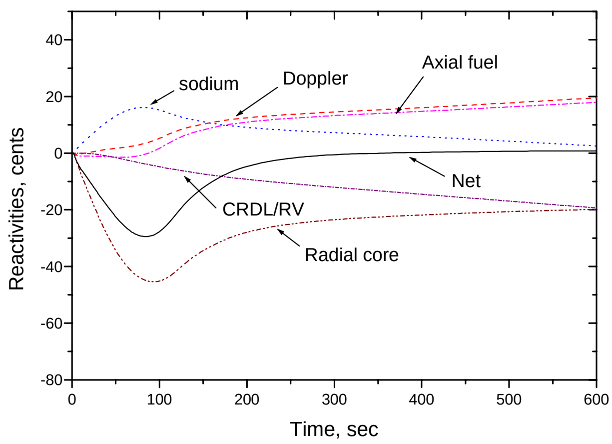

Fig. 9 Reactivity changes during the ULOF event

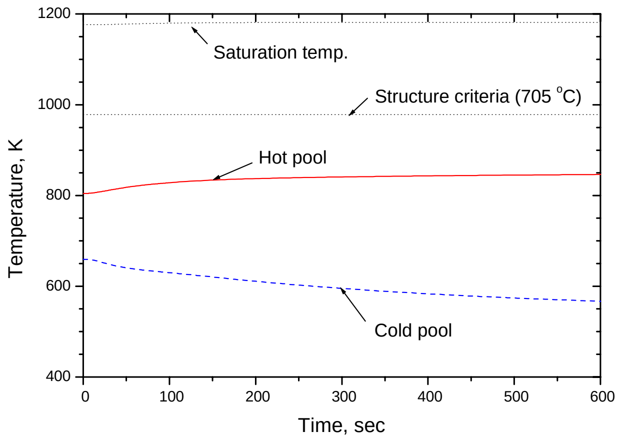

Fig. 10 Hot and cold pool temperatures during the ULOF event

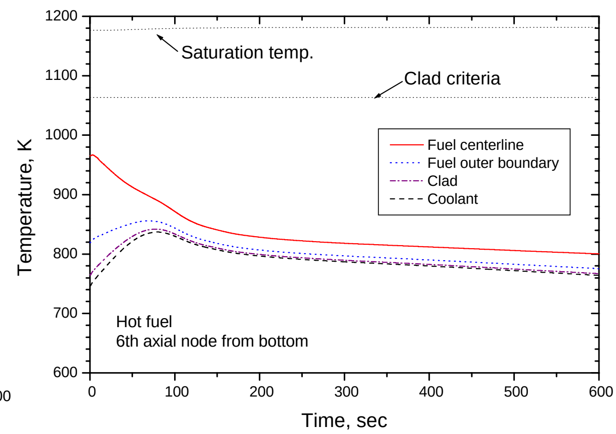

Fig. 11 Fuel temperatures during the ULOF event (6th axial node)

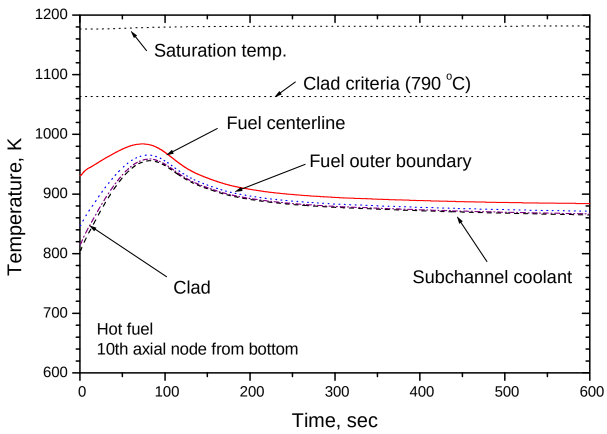

Fig. 12 Fuel temperatures during the ULOF event (10th axial node)

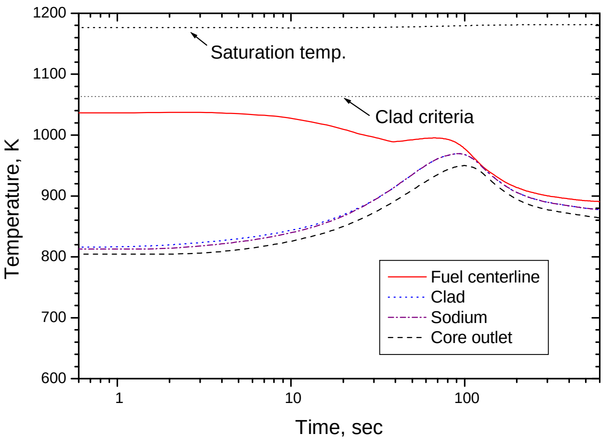

Fig. 13 Peak temperatures during the ULOF event

3.3 ULOF/LOHS

The accident of ULOF combined with the loss of heat sink (ULOF/LOHS) is more severe ATWS event than the ULOF. During this event the IHXs stop removing the heat from the primary heat transport system (PHTS) to the steam generators (SGs). The ULOF combined with LOHS is initiated from full power condition by the trip of the primary EM pumps followed by coastdown. The key parameters during the ULOF/LOHS are shown in Figs. 14 through 19. The net reactivity is stabilized well by the various feedback effects in 10 minutes as shown in Fig. 15. The fuel temperatures are also stabilized near to the sodium coolant temperature, which is maintained nearly constant after about 5 minute as a result of the reduced power due to reactivity feedback and the balance between power generation and heat removal through the passive vessel cooling system (PVCS) and PDRC.

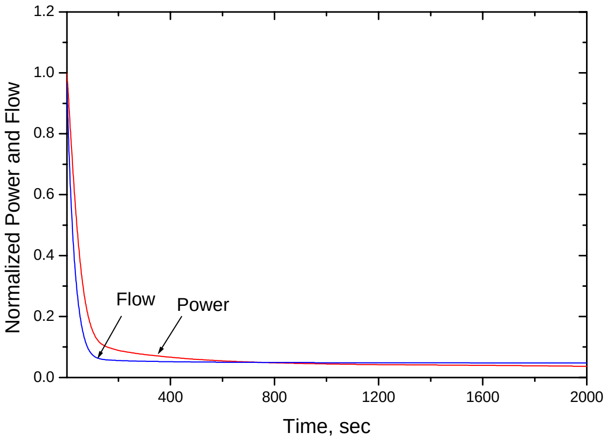

Fig. 14 Normalized power and flow (ULOF/LOHS)

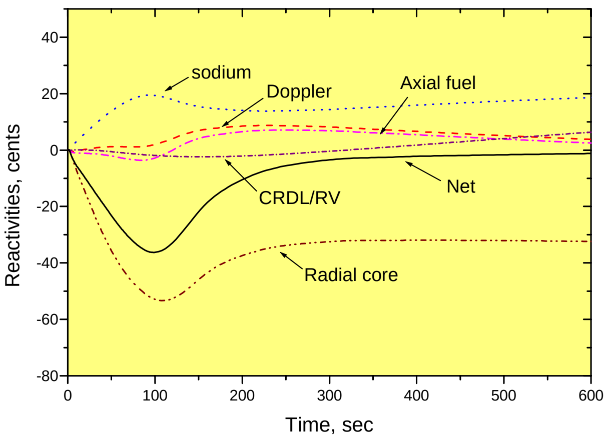

Fig. 15 Change of each reactivity components (ULOF/LOHS)

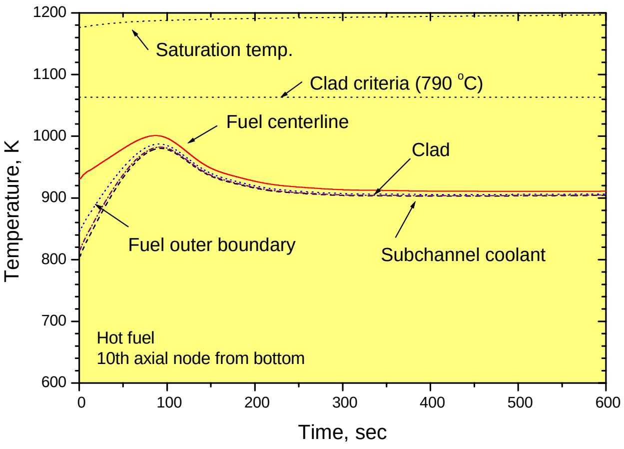

Fig. 16 Fuel temperature distribution at 10th axial node of hot fuel (ULOF/LOHS)

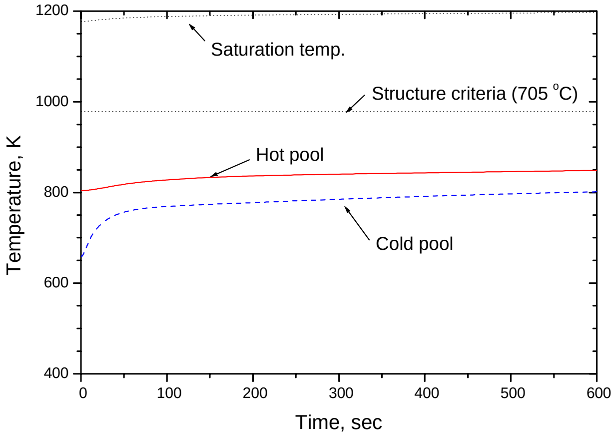

Fig. 17 Hot and cold pool temperatures (ULOF/LOHS)

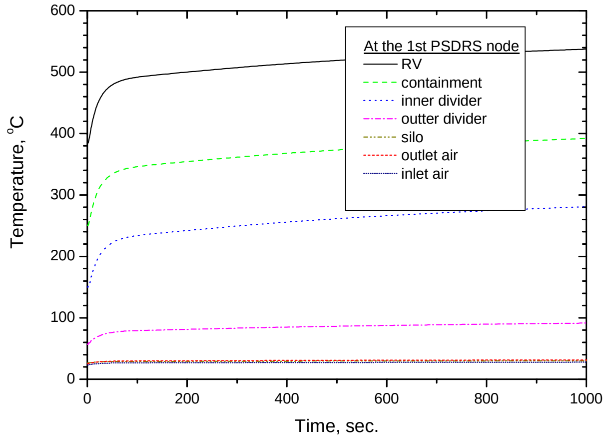

Fig. 18 Temperatures of wall and air in the PSDRS (ULOF/LOHS)

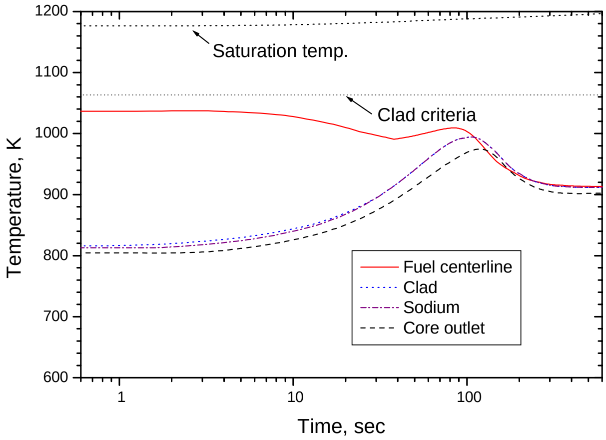

Fig. 19 Peak temperatures (ULOF/LOHS)

4. Conclusion

The safety of LMR could be threatened by the reactivity insertion or under-cooling events without the scram of reactor. Three events of ATWS, UTOP, ULOF, ULOF/LOHS are analyzed to evaluate the inherent safety of the newly designed Korean liquid metal-cooled reactor of 600 MW. The criteria for safety are the temperatures of fuel, clad, and coolant, which are determined by the combination of the reactivity feedback mechanisms. The fuel and structure temperature should be maintained within the temperatures ensuring the integrity of the materials. The coolant temperature also should be maintained below the boiling temperature. The safety criteria for fuel, clad, and coolant are 1343 K (1070 ^{o}C), 1063 K (790 ^{o}C), and 1343 K (1070 ^{o}C), respectively.

The peak fuel temperatures for the three typical ATWS events are well within the temperature limit of fuel. The clad and sodium temperatures are also do not exceed the safety criteria during the transients. Therefore, the inherent safety characteristics of the Korean 600 MWe LMR are clearly guaranteed by the results of the present analyses. Further, it is concluded that the evolutionary core design and other design efforts accommodate well the increased reactor power to guarantee the safety of the KALIMER-600 against the events of ATWS. More detailed analyses will follow along with future design progress.

Acknowledgements

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