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Analyses of Two Typical Pipe Breaks in Korea Advanced Liquid Metal Reactor

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Abstract

Two typical postulated breaks in the primary pump discharge pipe are analyzed to assure the inherent safety of Korea Advanced Liquid Metal Reactor. The amount of increase in the fuel and the coolant temperatures is the most important parameter in the analyses. The stabilization of the transient due to reactivity feedback is also important. It is assumed that one of the four pipes connecting the pump discharge to the core inlet plenum is broken. It is also assumed that the reactor is not scrammed after the initiation of the break, therefore, the pumps keep on running during the accident. The analysis is performed with SSC-K code, which was developed for the analysis of the transient system response of a pool-type reactor. As soon as the break occurs, the core flow decreases drastically to 65 % of full flow in the base case. A more conservative case is also analyzed in which the core flow is reduced artificially to 50 % of full flow. The reactor power stabilizes by reactivity feedback effects in about 10 minutes. The increase of the fuel and coolant temperatures due to the sudden reduction of the core flow are also mitigated with a large margin to coolant saturation temperature. The gas expansion module plays an important role to provide the dominant reactivity feedback when the core flow is reduced less than 50% of full power. It is concluded that a sufficient subcooling margin is maintained to guarantee the inherent safety of KALIMER against a pipe break.

1. Introduction

Korea Advanced Liquid Metal Reactor (KALIMER) is a pool-type liquid metal-cooled reactor having four intermediate heat exchangers (IHXs), four electromagnetic-type primary coolant pumps. In the KALIMER conceptual design [1], focus has been made on the nuclear steam supply system (NSSS) and essential BOP (Balance of Plant) systems. The ultimate objectives for KALIMER conceptual design are to make it safer, more economical, more resistant to nuclear proliferation, and yield less impact on the environment. KALIMER has a net electrical rating of 150MWe and the required core thermal output is 392 MWth.

The primary heat transport system (PHTS) of KALIMER mainly delivers the core heat to the intermediate heat transport system (IHTS) and IHTS works as the intermediate system between PHTS and the steam generator system (SGS) where the heat is converted to steam. The steam generator is a once-through type with helical tubes generating superheated steam. The reactor core, the primary coolant pumps and the intermediate heat exchangers (IHXs) are immersed in a large volume of sodium in the primary pool as shown in Fig.1. A vertical wall, called reactor baffle, divides the primary pool into hot and cold pools. The large thermal inertia provided by the large pool enhances the plant safety. The IHTS consists of two loops and each loop has its own steam generator and related systems.

The system reliability is improved by using electro-magnetic (EM) pumps that have no moving parts for both of the primary and intermediate coolant pumping. The flow inertia device compensates for the low momentum inertia of the EM pump. The device stores rotating kinetic energy when the EM pump runs normally but supplies electricity to the EM pump by converting the stored rotating kinetic energy to electricity at pump power supply failure. The primary EM pumps transfer the cold sodium in the cold pool into the core through the inlet plenum. Elevation differences feed the hot sodium in the hot pool into the inlet of the IHXs, past the tube bundle and into the cold pool. The only primary piping is from the discharge side of the pump to the core inlet plenum.

The use of a passive mechanism has, in general, superior reliability in mitigating an accident. In addition, long grace time at an accident provides improved reliability of the plant safety function and more flexibility in coping with an accident. The safety systems of KALIMER are based on the enhanced safety features such as using metallic fuel, the ultimate shutdown system (USS), the gas expansion module (GEM) and the passive safety decay heat removal system (PSDRS), which improve the reliability of KALIMER safety function. The large thermal capacity of the pool provides more time to cope with abnormal events and higher probability to terminate the abnormal events before their entering into accidents. KALIMER accommodates unprotected anticipated transients without scram (ATWS) events without operator action, and without the support of active shutdown, shutdown heat removal, or any automatic system without damage to the plant and without jeopardizing public safety. Neither operator action nor offsite support is required for at least three days without violating core protection limits in an accident.

The KALIMER design highly emphasizes the inherent safety, which maintains the core power reactivity coefficient to be negative during all modes of the plant status and under accidental conditions as well. The reactivity feedback mechanisms consist of Doppler, thermal expansion of the fuel and coolant, thermal bowing of the core, thermal expansion of the core structure and core support structure, and thermal expansion of the control rod driveline. These effects result from either the law of nature, or both the law of nature and core design.

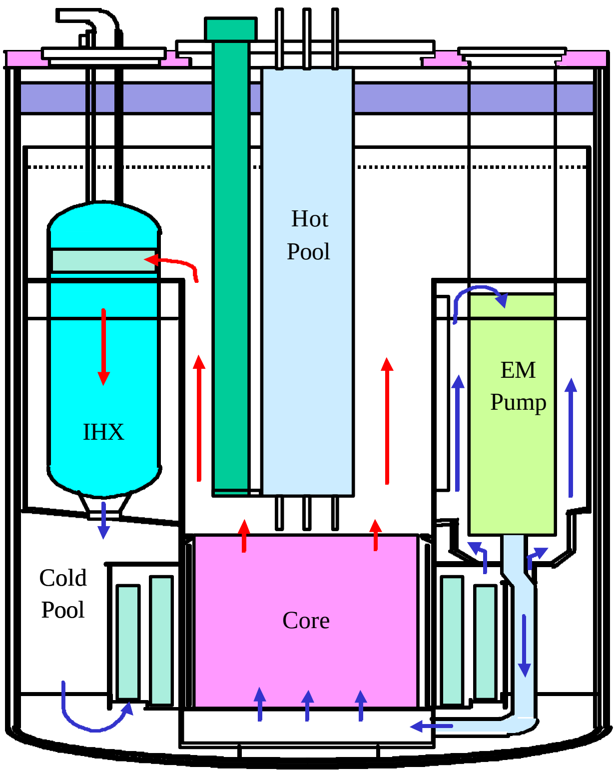

Fig. 1 Schematic of Primary Heat Transport System

The inherent safety characteristic against postulated events is the most remarkable superiority of a liquid metal cooled reactor (LMR) to other types of reactor. One of the major threats to the safety of LMR is a loss of flow event accompanied with a failure of the reactor shutdown system. This situation is usually referred to as an unprotected loss of flow (ULOF). The inherent safety of KALIMER during the ULOF [2] has been assessed for the situation of all pump trips followed by coastdown. It was assumed that the decay heat was removed by four intermediate heat exchangers (IHXs) and the safety grade system of the passive safety decay heat removal system (PSDRS). The results showed that the power was stabilized by the reactivity feedback of the system even though the effect of the gas expansion module (GEM) was not taken into account.

One of the possible mechanisms of loss of flow is the seizure of one or more pumps and the rupture of a primary pipe [3]. In a loss-of-flow-type accident, the power-to-flow ratio is the key parameter that determines the consequences of the accident. Thus, the initial pump behavior plays an important role for the plant safety when the pump is still running. On the other hand, more severe mismatch between the power and flow is possible in a pump seizure or a pipe break accident because the core flow drops more abruptly. Therefore, it is required to analyze the loss of flow due to the break of inlet pipe connecting the pump discharge and core inlet plenum.

The present study analyzes a postulated break in the primary pump discharge pipe to assure the inherent safety of KALIMER. KALIMER is a pool-type liquid metal sodium cooled fast reactor plant. The main concern of the accident includes the amount of subcooling margin reduction, i.e., the degree of increase in the fuel and the coolant temperatures. The stabilization of power associated with reactivity feedback is also an important aspect of the accident. The analysis is performed with the SSC-K code, which is developed based on SSC-L code for the analysis of the transient system response. Actually, the possibility of sodium loss by pipe break is very low and the large thermal capacity of the pool makes the system transient slower.

2. SSC-K Code Models

The SSC-K code [4] 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. The SSC-K is based on the methods and models of SSC-L [5], which was originally developed to analyze loop-type liquid metal reactor transients. Because of the inherent difference between the pool and loop designs, major modification to the SSC-L has been made in order to analyze the thermal hydraulic behavior within the pool-type reactor. Now, the SSC-K code has the capability to analyze both loop and pool type liquid metal cooled reactors. Additional developments in the SSC-K code include models for reactivity feedback effects for the metallic fuel, and the PSDRS. Also a two dimensional hot pool model has also been employed into SSC-K for analyzing the thermal stratification phenomenon in the hot pool. The control system model in SSC-K is flexible enough to handle any control system.

The SSC-K code was used for assessment of the inherent safety features in the KALIMER conceptual design. 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 is currently being used as the main tool for system transient analysis in the KALIMER development. A full plant model for SSC-K is used to represent KALIMER as shown in Fig. 2 in which several major components are represented. The PHTS is represented by the flow passage in the pool, the primary pump, and the shell side of the IHX. The IHTS consists of the tube side of the IHX, the connecting pipes, the shell side of the steam generator (SG), and the intermediate pump.

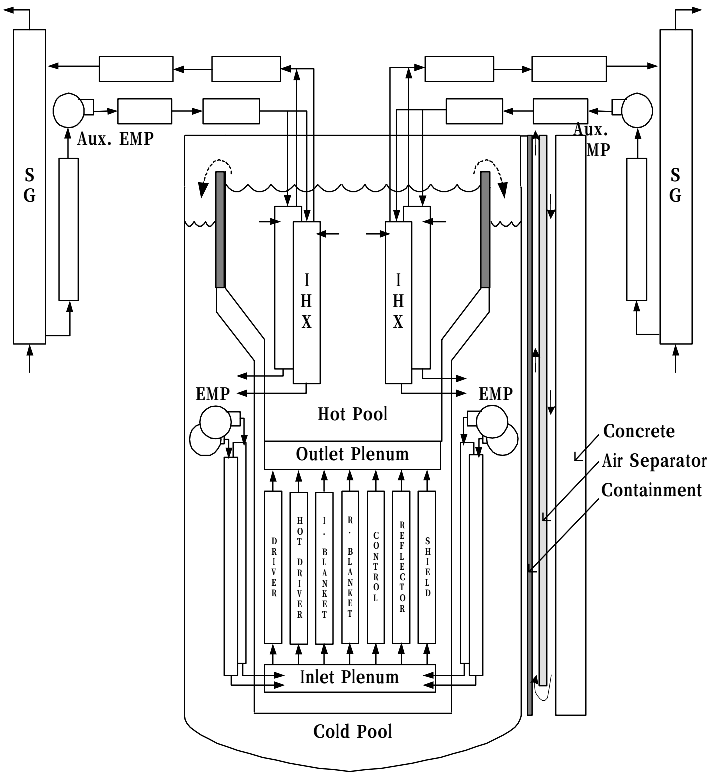

Fig. 2 SSC-K Modeling for KALIMER Plant

2.1 Pool Thermal-hydraulic Model

A major modification of SSC-K has been made in order to analyze the thermal hydraulic behavior within the pool. In KALIMER, both the hot and cold pools have free surfaces and there is direct mixing of the coolant with these open pools prior to entering the next component. Therefore, at least two different flows would have to be modeled to characterize the coolant dynamics of the primary system. The first flow from the pump to the hot pool through the core would respond to the pump head and losses in the flow passages.

<u>Pump Flow</u>

$$\frac{dW_p(k)}{dt}\sum_p \frac{L(k)}{A(k)} = P_{Po}(k) - P_{Rin} - \sum_p \Delta P_{f,g}(k), \qquad k = 1, \ldots, N_{path} \tag{1}$$

In above equation, the pump exit pressure, P_{Po}, is obtained from

$$P_{Po} = P_{Pin} + \rho_{Pin} g H \tag{2}$$

where H is the pump head, obtained from the pump characteristics. The pump inlet pressure can be obtained by calculating the elevation head for the cold pool sodium level.

The other flow from IHX to the cold pool would respond to the level difference between the two pools as well as the gravity gain in the IHX because the gravity gain could be significant for low-flow conditions.

<u>IHX Flow</u>

$$\frac{dW_{1X}(k)}{dt}\sum_{X}\frac{L(k)}{A(k)} = P_{Xin} - P_{Xo} - \sum_{X}\Delta P_{f,g}(k), \qquad k = 1,\Lambda\ , N_{path} \tag{3}$$

The IHX inlet and exit pressures, P_{Xin} and P_{Xo}, are obtained from static balance as

$$P_{Xin} = P_{gas} + r_{h} g(Z_{HP} - Z_{Xin}) \tag{4}$$

$$P_{Xo} = P_{gas} + r_{c} g(Z_{CP} - Z_{Xo}) \tag{5}$$

The equations are solved coupled with the differential equations derived by mass and momentum conservation at the core inlet plenum.

When reactor scram occurs, the heat generation is reduced almost instantaneously while the coolant flow rate follows the pump coastdown. This can result in a situation where the core flow is colder than the bulk hot pool sodium. This temperature difference leads to stratification when the flow momentum is not large enough to overcome the negative buoyancy force. The two-zone model employed in the original SSC-L code has been modified. The hot pool is divided into two perfectly mixing zones determined by the maximum penetration distance of the core flow. The time rate change of energy in the pool is added to energy balance equations in the SSC-K code to provide conservation. In addition, the two-dimensional pool model has been developed to calculate the coolant temperature and velocity profiles in the hot pool. The governing equations for conservation of mass, momentum, energy, and both turbulent kinetic energy and the rate of turbulent kinetic energy dissipation for the ê-å turbulence model are made in a generalized coordinate system. The SIMPLEC algorithm is used for pressure-velocity coupling. After validation of the stand-alone version of the two-dimensional pool model [6] against the sample problem, it is coupled into the SSC-K code.

2.2 Reactivity models for a metallic fueled core

To facilitate modeling of the metal fuel used in KALIMER, several reactivity models are modified in the SSC-K code. For neutronic calculations, SSC-K uses point kinetic equations with detailed reactivity feedback from each channel. Reactivity effects are required both for transient safety analysis and for control requirements during normal operation. Reactivity changes are calculated for control rod scram, the Doppler effect in the fuel, sodium voiding or density changes, fuel thermal expansion, core radial expansion, thermal expansion of control rod drives, and vessel wall thermal expansion. Figure 3 shows the components of reactivity feedback considered in the KALIMER core. The effect of fuel expansion becomes more significant when metallic fuel is used.

In addition to the reactivity model, a GEM model has been developed for SSC-K. The GEM assemblies are added to KALIMER core in order to supplement the negative reactivity feedback once the pump is tripped. For safety margin in the event of loss of the primary coolant flow, GEMs are included at the periphery of the active core. A GEM has the same external size and configuration as the ducts of the other core assemblies. The GEMs are hollow assembly ducts, which are open to flow at the bottom but are closed to flow at the top. The GEMs are filled with vessel cover gas before insertion into the core, and this gas is compressed as the GEMs are filled with sodium.

With the primary pumps on, the high pressure in the inlet plenum compresses the gas captured in the GEMs and raises the sodium level in the GEMs to a region above the active core. When pumping power is lost in the primary system and the pressure drops, the gas expands, which results in displacing the sodium in the GEMs to a level below the active core. The resultant void near the core periphery increases neutron leakage and introduces significant negative reactivity, which limits the peak temperatures attained during the loss of flow events. Currently, the sodium density inside the GEM is assumed to be the axial average of the neighboring channels. A sensitivity study is needed to investigate the effect of sodium density on the sodium level. The temperature of the GEM gas is assumed to be the average of the structural temperature of the neighboring channels.

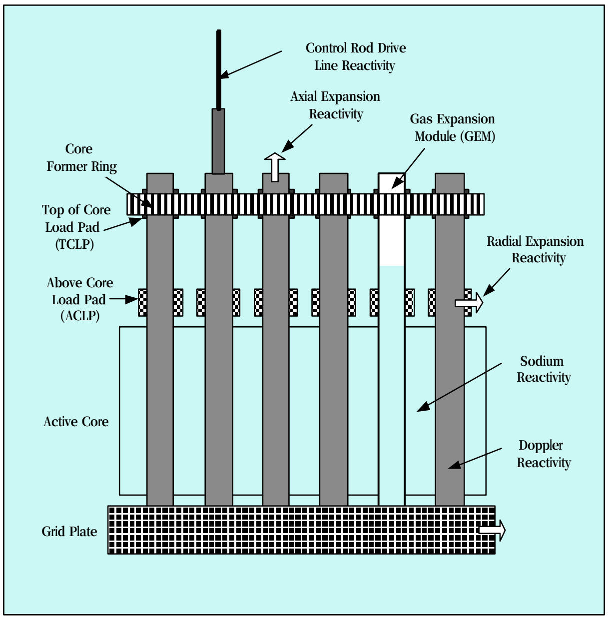

Fig. 3 Reactivity Components in a Metallic Fueled Core

3. Analysis Method

In the KALIMER design, the postulated pipe rupture can only happen in pump discharge line to reactor core. This accident reduces the core mass flow and thus may increase the fuel temperature as well as the coolant temperature. The main concern of the accident is associated with reduction of the subcooling margin and power stabilization resulting from reactivity feedback. In the present analysis it is assumed that one of the four pipes connecting the pump discharge to core inlet plenum is broken. The break is located at 3.7 m below the pump outlet and the diameter of the break is 0.4 m, therefore, the break is about 0.1257 m^2. It is also assumed that the reactor is not scrammed after the initiation of the break, therefore, the pumps keep on running during the accident. The break is assumed to occur 5 seconds after the transient.

To describe the broken path, Eq. (1) has to be modified to:

$$\frac{dW_p}{dt}\sum_{uob}\frac{L}{A} = P_{Po} - P_{bin} - \sum_{uob}\Delta P_{f,g} \tag{6}$$

When a pipe break occurs, an additional equation is needed to describe the flow downsteam of the break:

$$\frac{dW_{dob}}{dt}\sum_{dob}\frac{L}{A} = P_{bo} - P_{Rin} - \sum_{dob}\Delta P_{f,g} \tag{7}$$

The inlet and outlet pressures at break location, P_{bin} and P_{bo}, respectively, are calculated by break model. The external pressure for the break, which is needed to compute these pressures, is obtained from static balance as

$$P_{ext} = P_{gas} + r_C g(Z_{CP} - Z_b) \quad (8)$$

The external pressure for the break corresponds to the static head of the cold pool. This pressure acts as the opposing pressure against the flow out of the break. The value of this pressure is much larger than that for the loop-type design, which is generally equal to atmospheric pressure until the sodium in guard vessel covers the break location in a loop-type reactor. This will make the pipe break in pool-type designs less sever relative to loop-type designs. The break model in SSC-K is basically the same as that in SSC-L.

The present version of the SSC-K code allows the modeling of a primary system with only one loop. In other words, the flow paths constituted by the four discharge lines and four inlet pipes are simplified into one imaginary flow path. Therefore, there is the possiblity that the break flow from one inlet pipe is not modeled accurately and the flow from the intact sides to the broken side is not described physically. To take into account this deficiency in the break modeling, a sensitivity study is performed in the present study to adjust the resultant core flow. The base case with the break area equivalent to one pipe diameter results in the reduction of core flow about to 65% of full flow. The break flow is artificially increased to reduce the core flow below 50% of full flow, which compensates conservatively the deficiency in primary system modeling. The summary of reactivity worth for KALIMER breakeven core is provided in Table 1.

Table 1. Reactivity worth for KALIMER breakeven core

Reactivity Parameter	**BOEC**	**EOEC**
Fuel Temperature(Doppler) Coefficient (d rho/ dT)		
Sodium Flooded	$-0.08692T^{-1.44}$	$-0.08191\ T^{-1.42}$
Sodium Voided	$-0.08787T^{-1.47}$	$-0.08657\ T^{-1.46}$
Uniform Raidal Expansion Coefficient		
(dk/k) /(R/dR)(pcm/%)	-143	-141
dk/dT (x 10^{-4})(1/K)	-8.6899	-8.3742
Sodium Void Effect (pcm)		
Driver Fuel (DF)	560.63	760.40
Internal Blanket (IB)	606.71	664.46
Radial Blanket (RB)	-186.66	-159.50
DF + IB	1205.75	1462.89
DF + IB + RB	1012.74	1298.46
DF + IB + RB + GEM	-302.48	130.54
Control Rods (pcm)		
1 Rod	970.13	1021.63
3 Rods (Cluster)	3632.25	3793.89
6 Rods (Total)	7688.38	8051.33
Interaction Factor		
Adjacent Rods	0.925	0.931
Clusters	1.058	1.061
GEM (pcm)	1228.55	1086.66
USS (pcm)	1441.87	1847.07

4. Results and Discussion

The analysis is performed using the SSC-K code. As soon as the break occurs, all flow rates are changed abruptly as shown in Fig. 4. In the base case, the core flow decreases drastically to less than 65% the full flow within 5 seconds after the break. The core flow is reduced because the pressure at the break after the initiation of the transient is changed rapidly from steady state value in the pipe to the hydraulic head established by the sodium in the cold pool. The pump discharge flow increases to about 146% of the initial steady value. All coolant passing through the pumps does not enter the core and part of it, 56% of the total pump flow, discharges through the break into the cold pool.

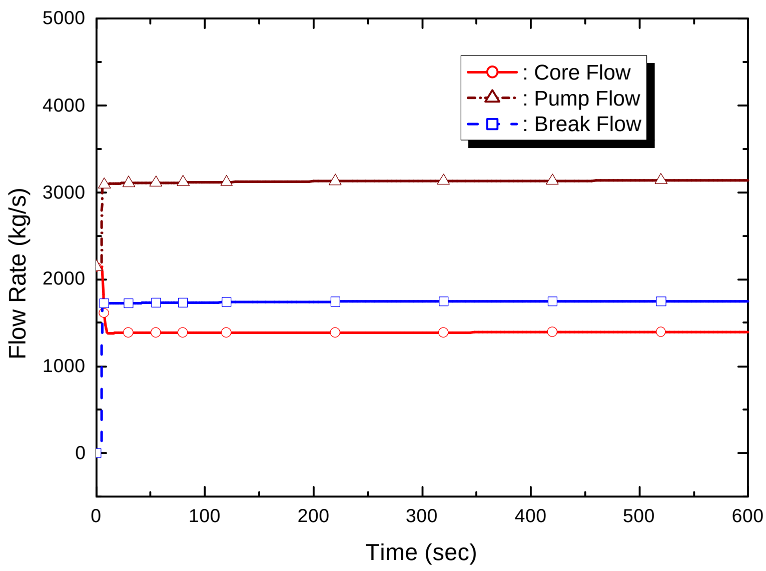

Fig. 4 The core, pump and break flow in pipe break event (Base Case)

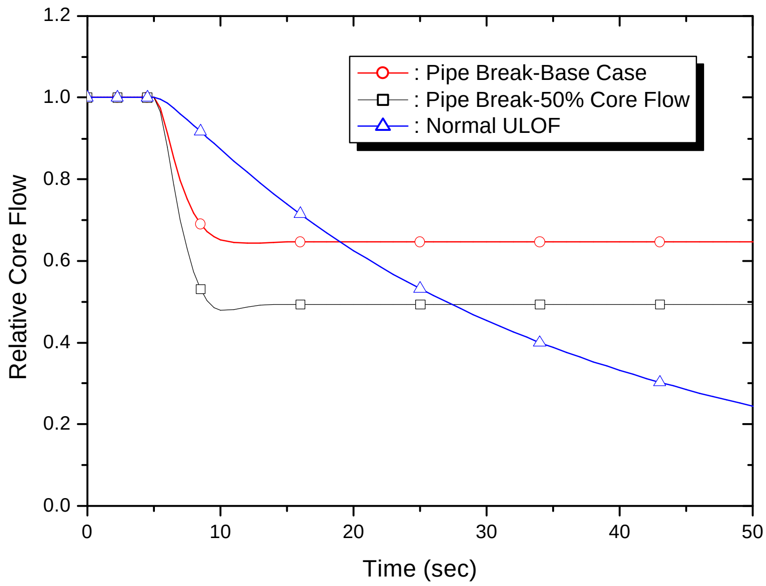

Fig. 5 The core flow rate in the pipe rupture events and the normal LOF

In Fig.5, the core flow rates in cases where a pipe breaks are compared with the flow rate in the normal LOF case. The core flow in pipe break case drops more rapidly than that of the normal LOF case, thus, more severe mismatch between the power and the flow can be induced. The relative power and flow during the transients are trended in Fig. 6 for the base case and for the 50% core flow case, respectively. For the base case the reduction in core flow is much larger than the reduction in power. Therefore, the transient is expected to accompany higher coolant and fuel temperatures.

On the other hand, the power generation rate is reduced much more than the core flow rate in the 50% core flow case. The reason for these power trends can be found in the reactivity feedback effects as shown in Figs. 7 and 8. The most remarkable difference is the effect of GEM. KALIMER is equipped with the advanced safety feature of GEM to provide additional negative reactivity in response to loss-of-flow events. When the pumps are operating at normal condition, sodium is pumped into the GEM, and the trapped helium gas is compressed into the region above the active core. However, when the pumps are off or the flow to core is reduced, the helium gas region expands into the active core region, displacing the sodium in the GEM below the active core top. The resultant void near the core periphery increases neutron leakage and introduces significant negative reactivity.

For the adjusted 50% core flow case, the effect of GEM is dominant and all the feedback effects except the sodium reactivity and the GEM reactivity go positive. The GEM level for the base case remains above the active core top as shown in Fig. 9 and the negative reactivity is not provided by GEM. In contrast, the GEM level for the 50% core flow case is maintained below the active core top and the negative reactivity is inserted by the effect of the GEM.

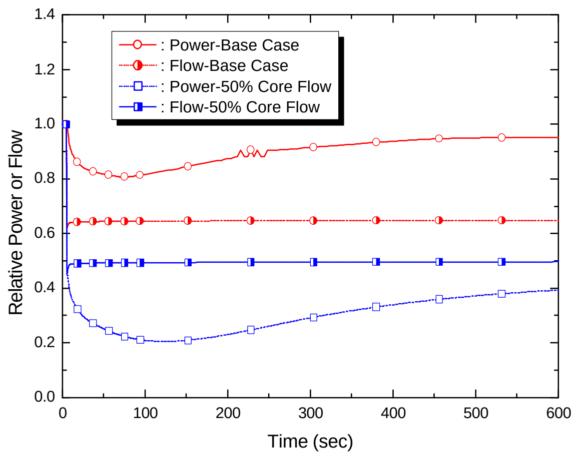

Fig. 6 Trends of relative power and flow in pipe rupture events

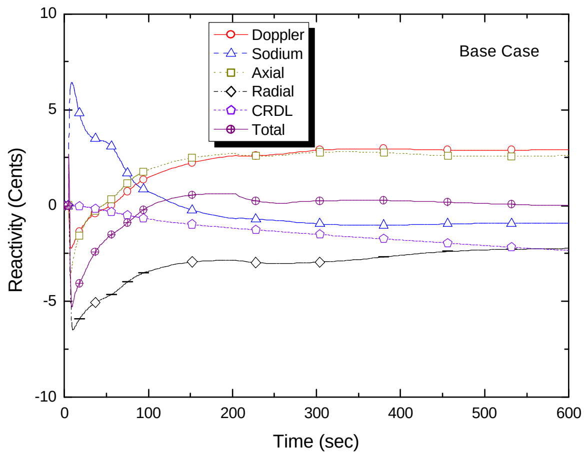

Fig. 7 The reactivity feedbacks in a pipe rupture event (Base Case)

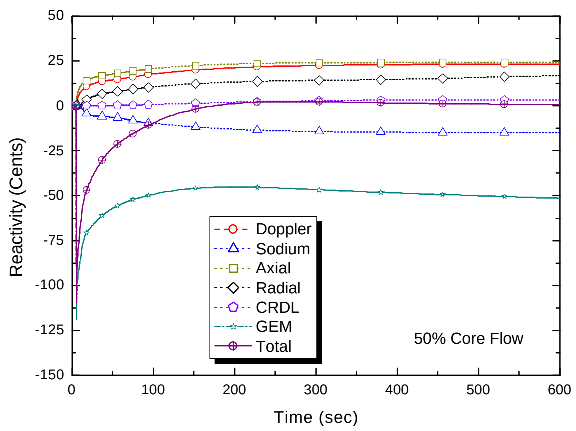

Fig. 8 The reactivity feedbacks in a pipe rupture event (Adjusted 50% core flow)

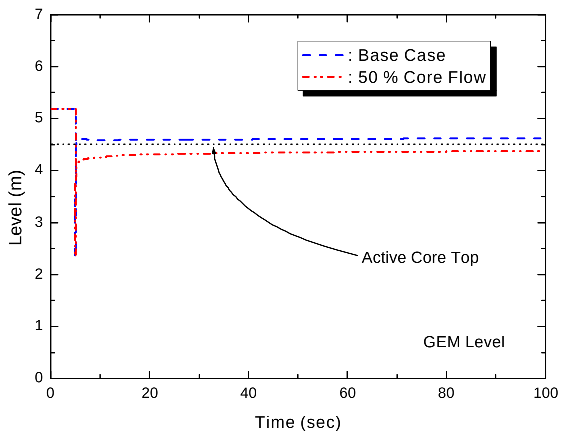

Fig. 9 Prediction of GEM levels in pipe rupture events

The GEM level is influenced by the pump inertia force exerted to the inlet plenum and the level difference between the hot pool and the cold pool. The hot and cold pool levels are predicted for the base case and the 50% core flow case in Fig. 10. The pressure at the break rapidly drops from the initial value to that corresponding to the sodium head in the cold pool, so that the cold pool level soars by more than 3 m for the base case and 4 m for the 50% core flow case instantaneously. It leads to the decrease of the pressure drop between the core inlet and the core outlet, thus results in the core flow reduction. The cold pool level, thereafter, eventually remains nearly constant under new condition established. The hot pool level also keeps almost initial value except negligible reduction at the very beginning.

The cold pool temperature begins to decrease by the addition of the cold coolant flowing out from the pump outlet and finally reaches an equilibrium temperature at about 400 seconds in the base case of pipe break. The hot pool temperature increases until 100 seconds due to the reduction of core flow and the power-to-flow mismatch and then an equilibrium temperature is maintained after that. Its change rate is rather slow and small due to the large heat capacity of the pool. The pool temperatures for the 50% core flow case show quite different behaviors from those of the base case. Because the rate of power reduction is larger than the rate of flow reduction as shown in Fig. 8, both the hot and cold pool temperatures decrease continuously after the initiation of the break. The pool temperature behaviors can be seen in Fig. 11.

The fuel, cladding, and coolant temperatures for base case are represented in Fig. 12. The temperatures in the core channel increase drastically due to the rapid drop of the core flow, which leads to the large power-to-flow mismatch. The fuel centerline temperature shows the same trend with the power variation governed by the reactivity feedbacks. The clad and sodium temperatures show similar early peaks, however, they stabilize much earlier than the fuel centerline temperature. The peak fuel temperature is about 1109 K and the peak sodium temperature is about 865 K, which guarantees the subcooling margin of more than 400 K. The fuel and coolant temperatures in the core for the 50% core flow case are not detailed because it is presumed that there will be no remarkable temperature increase in this case.

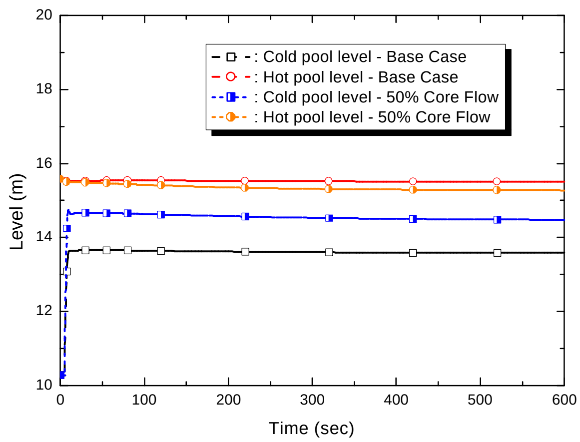

Fig. 10 Hot and cold pool levels in pipe rupture events

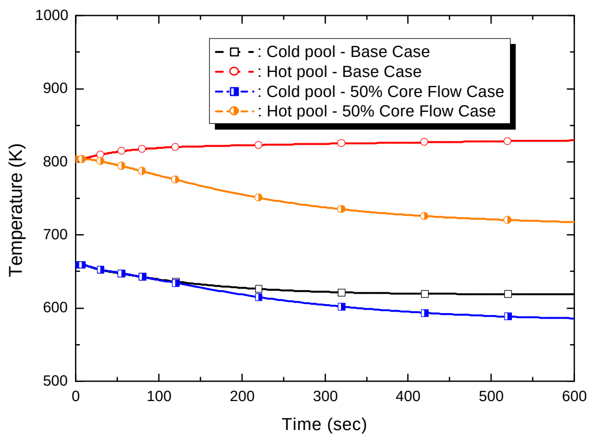

Fig. 11 Hot and cold pool temperatures in pipe rupture events

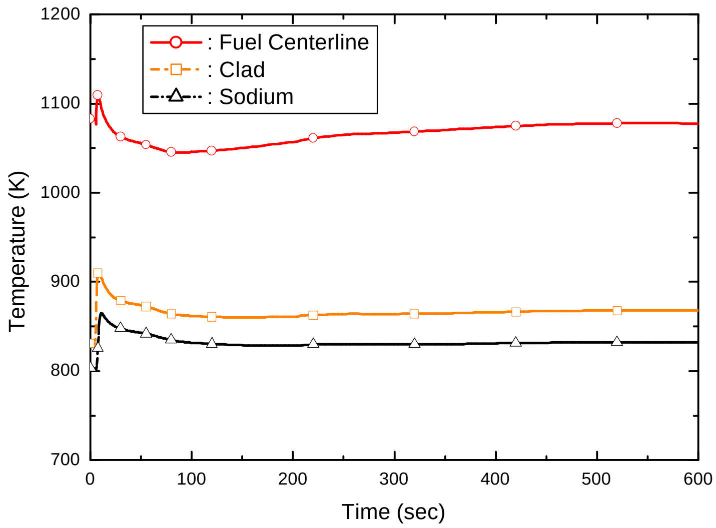

Fig. 12 Fuel and coolant temperatures in a pipe rupture (Base case)

5. Conclusion

The break of one of the four core inlet pipes of KALIMER is analyzed and the inherent safety of KALIMER against a pipe break is evaluated in the present study. The reactivity feedbacks, the power trend and other parameters are predicted to show quite different behaviors depending on whether the GEM becomes effective or not. In the base case, in which the break area is equivalent to the diameter of inlet pipe, the core flow is reduced to about 65% of full flow and the GEM level remains above the active core top. In this case, the power is governed by the combined effect of the reactivity feedbacks induced by the Doppler, sodium, radial, axial and control rod driveline behaviors. To reinforce the deficiency in modeling the pipe break with SSC-K code, the core flow is artificially reduced to about 50% of full flow, which results in the operation of GEM.

Even though the fuel and sodium temperatures increase after the initiation of the breaks, more than 400 K of subcooling margin is always guaranteed for the pipe break accident in KALIMER. In addition, the power stabilizes by the reactivity feedbacks even though the GEM is not effective when there is smaller reduction of core flow. If the core flow is reduced enough to drop the GEM level below the active core top, the negative reactivity of GEM results in rapid drop of reactor power and guarantees enough subcooling margin and stabilization of the reactor dynamics.

The present study ensures the superior resistance and inherent safety of KALIMER against the pipe break accident. The slow response of the hot pool temperature, which is another advantage of pool type liquid metal reactor, is also demonstrated. The GEM is found to be very helpful to provide sufficient negative reactivity for the passive shutdown mechanism and to mitigate the consequences of some spectrums of pipe breaks.

Acknowledgements

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