Development of a new SSC-K version for the Lead-cooled Liquid Metal Reactor Analysis

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1. Introduction

The current version of SSC-K which is being used for the KALIMER safety analysis, was initially designed for a sodium cooled, pool type liquid meter reactor (LMR). These days, however, interest has risen in the lead cooled liquid metal reactor world wide as an option for the generation IV reactors.[1] The stability of lead on a chemical as well as its nuclear interactions is a key merit of this option. Amid such an effort, the analysis capability of the lead cooled reactor needs to be prepared for its potential development.

To meet the necessity, this study aims at modifying parts of the SSC-K programs for its application to the lead cooled reactor analysis.

2. Methods and Results

The first step may be the replacement of the sodium physical properties with those of lead in SSC-K to represent the lead coolant characteristics. The core data associated with the reactivity feedbacks should also be provided.

2.1 Modification of the Physical Properties [3]

A scope of the study is to modify the subroutines in SSC-K for the lead coolant properties. The development of the other requirements such as the appropriate correlations for the lead coolant is beyond the present scope.

All thermodynamic properties in SSC-K are expressed in polynomials with the coolant temperature. Similar polynomials are used for the sodium coolant. For example, the enthalpy of lead is given by;

$$\begin{aligned} ENTHBH = {} & a_1/(T_C - T) + a_2 * ALOG(T_C - T) + a_3 * T \\ & + a_4 * (T_C - T) * (T_C - T) + a_5 * (T_C - T) * (T_C - T) * (T_C - T) \end{aligned}$$

(1)

where, T_c corresponds to the lead critical temperature (=5,400 K).

To find the temperature at a given enthalpy, the Newton method has been used with the convergence criterion of 10^{-12}. Although the primary loop may be cooled by lead, the intermediate loop coolant is still assumed to be sodium at the present time. Therefore, the sodium properties in the subroutines associated with the primary loop have been modified, while those in the intermediate loop remain unchanged.

2.2 Determination of the Steady-state Conditions

2.2.1 Coolant Flow and Core □T

The feed-water and the steam dome boundary conditions in a steam generator should be modified in the input to account for the coolant change. Since the detailed design has not been completed yet, the lead coolant flow is calculated under the assumption that the same □T as that of the sodium core could be maintained in the core.

The lead flow is a simply set of 10 times the original sodium flow, tentatively. The estimation is based on the density ratio between the two coolants. Then, the steady state lead flow is obtained by a running null transient calculation for 600 sec.

2.2.2 Pressure Drop Calculation

The pressure drop for lead may be underestimated, if that for sodium is employed as it is now. The main source of the disagreement comes from the density difference between the two coolants. The frictional pressure drop and form loss will roughly be proportional to the density ratio, under the assumption that the coefficients associated with the frictional pressure drop and form loss remain constant, as seen in the frictional pressure, for instance, given by;

$$\Delta P_f = \frac{f \square L}{2 \square D_H} \frac{\dot{m}^2}{(\rho \square A_c)}$$

(2)

Therefore, the pressure drop by the lead coolant is multi-plied with the density ratio as follows.

$$\Delta P_{pb} = \Delta P_{NA} \square (\frac{\rho_{pb}}{\rho_{NA}})$$

(3)

2.3 Transient Calculation

The previously obtained results [4] for the KALIMER safety analyses are used to evaluate the present SSC-K lead version. The results calculated under the same conditions for TOP(Transient Over-Power), LOF (Loss of Flow), and ULOHS(Loss Of Heat Sink) are compared. These events are selected instead of the unprotected cases, because the appropriate reactivity feedback models are not available in the lead version.

2.3.1 TOP

After a null transient calculation for 200 sec, a total 30 cents of positive reactivity is inserted for 15 sec. at the beginning of the transient. Figure 1 is a comparison of the pool temperatures for the two coolants. The lead temperature drops more slowly because its time constant is larger.

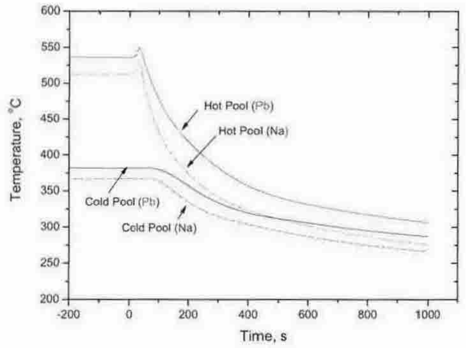

Fig. 1Comparison of the Pool Temp. (TOP)

2.3.2 LOF

The primary pumps are manually tripped at 0 sec and the reactor trip is assumed to occur at 20 sec after the transient initiation. The behaviors of both coolants show almost the same at both powers and flow rates. As represented in Fig. 2, the fuel, cladding, and the core coolant temperatures do not indicate a remarkable difference, however, the core coolant average temperature for lead is calculated a little higher because of its higher heat capacity as well as its higher heat transfer coefficient.

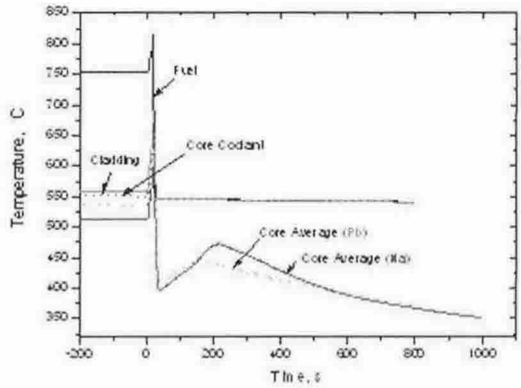

Fig. 2 Results of the temperatures in the core (LOF)

2.3.3 LOHS

After a LOHS accident, the reactor trip is assumed to occur at 20 sec. Fig. 3 represents the result of the pool temperature calculations. The pool temperatures for lead result in a faster growth than those of sodium, again due to the higher heat capacity as well as the higher heat transfer coefficient for lead.

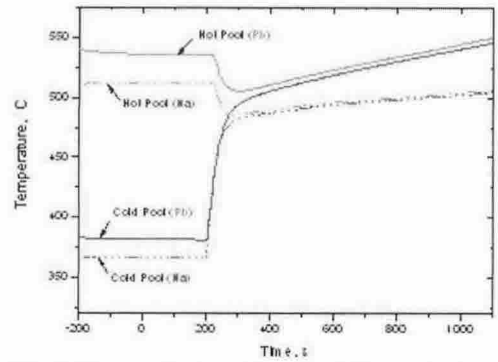

Fig 3 Comparisons of the Pool Temp. (LOHS)

3. Conclusions

The SSC-K lead version is evaluated by comparing the analysis results with those of the sodium version against TOP, LOF, LOHS accidents. The results are acceptable, but the extent of the evaluation is so limited that it may not be applied to the lead cooled reactor directly.

To accomplish this, other things like the core reactivity feedbacks and physical correlations corresponding to the lead coolant should also be modified properly and then some additional verifications must also be undertaken in the future.

Acknowledgements

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[3]Y.B. Lee, et al., "The Development of Technologies of Safety Analysis for LMR," KAERI/TR-2736/2004