ATWS Analysis with SSC-K for the new conceptual design of KALIMER-600

W.P.Chang, G.S.Ha, H.Y.Jeong, S. Heo, E.K. Kim, Y.B. Lee
Fluid Engineering Research Division, Korea Atomic Energy Research Institute, P.O.B ox 105, Yusung, Daejeon , Korea

1. Introduction

A new conceptual design of a 600 MW(e) LMR(Liquid Metal Reactor) following up the KALIER-150 [1] is categorized as an advanced reactor, which needs stricter safety requirements than a conventional one. The reactor accidents with a frequency over than 10^{-6} have been classified as severe accidents which are beyond the design-basis-accidents. Necessary safety measures of the advanced reactor, however, have been taken into account during its design stage by handling the accidents as ones equivalent to the design-basis-accidents despite their extremely low frequency. The rationale based on the fact that those accidents have a potential of either a considerable amount of radioactivity release, or power excursion by a large reactivity insertion. The representatives of those accidents are ATWS's, where reactor trips are excluded in their safety analyses by assuming a failure of the actuation of the protective systems. The present analyses, therefore, aim at not only identifying the characteristics of the newly developed reactor, but assessing its inherent safety. The UTOP (Unprotected Transient Over Power), ULOF (Uprotected Loss of Flow), and ULOHS (Unprotected Loss Of Heat Sink) analyses are the main analyses in this study.

2. Accident Analysis

The changes on the reactor size, arrangement of the fuels, and reactivity characteristics are made as the KALIMER capacity increases. The previous passive safety residual system, PVCS (Passive Vessel Cooling System) is also replaced with a PDRC (Passive Decay heat Removal Circuit) for the KALIMER-600. A complete set of the design data required for the safety analyses has not been fully generated as yet. In this regard, some roughly estimated key design parameters as well as design data reasonably extrapolated from those of KALIMER-150 had to be used. The design data for the safety analysis, however, have gradually been updated over the last 3-years for the sake of KALIMER-600's by using own design data. A key safety analysis using its own design data has become partly possible, to some extent, as the design develops.

The purpose of the present anlayles is to find whether the safety parameters are guaranteed to satisfy the safety limits during the ATWS accidents. The inherent safety driven by the reactivity change must be a primary concern in the evaluation. The analysis results obtained from 2002 through to 2004 are compared with each other for the central parameters threatening its safety. The SSC-K [3] Version 1.2 was used for the calculation in 2002 and 2003. In contrast, the SSC-K Version 1.3 is applied to the analysis in 2004. Some notable design changes along with the applied analysis models are summarized in [3].

3. Analysis Results

The comparative study addresses the fulfillment of the safety limits against the peak temperatures of the fuel, cladding, and sodium. The accident scenarios for the analyses are well described in [3].

(1) UTOP

The UTOP accident is initiated by a reactivity insertion due to a control rod ejection by the failure of the control rod driving mechanism, or an inadvertent operator action. The primary and intermediate sodium flows are assumed to operate normally in the analyses. The accident consequences depend largely on the inherent safety regardless of the amount of positive reactivity insertion. Figures 1 and 2 present the calculation results of the power and core safety related temperatures, respectively. The core power gives a considerable difference, due to the changes of both the reactivity model and the fuel physical properties.

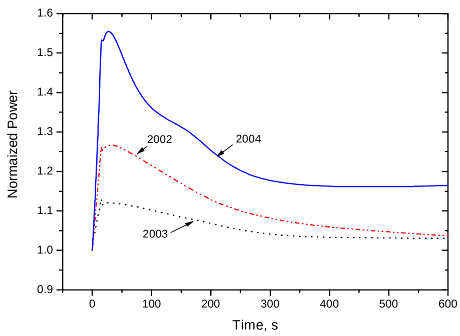

Figure 1 Core powers (UTOP)

(2) ULOF

The ULOF event is initiated by a loss of a normal flow

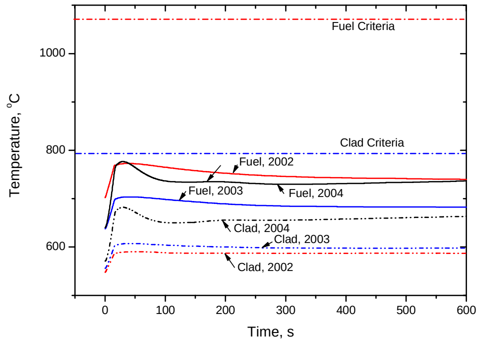

Figure 2 Fuel and cladding temperatures (UTOP)

The ULOF event is initiated by a loss of a normal flow into the core. There could be various types of a loss of flow. In the present analysis, however, it assumes that both of the two primary pumps stop all at once at 100 % of the core power and thereafter they begin a coast-down operation. The event could be caused by the loss of the off-site power, or a common failure of the pumps themselves. There may be a possibility that the fuel temperature does not fulfill the safety requirement unless a proper core flow is established. Figure 3 indicates that the requirement has been met according to the inherent reactivity feedback.

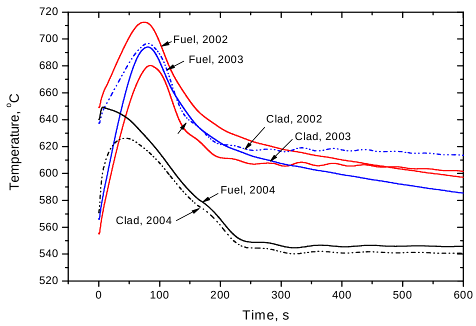

Figure 3 Fuel and cladding temperatures (ULOF)

(3) ULOHS

In order to suppress the propagation of the water-sodium reaction by a water leakage into the IHTS through the steam generator tubes, the IHX is to be isolated. The ULOHS safety analysis assumes no natural circulation within the IHTS due to a complete loss of its coolant. Consequently, it loses all its heat removal capability. The current pumps operation, however, is designed to be manually stopped at about 2,000 s after an accidents occurrence. The key point of the ULOHS analysis is to investigate whether the safety limit is guaranteed for 72 h without an operators' intervention.

In this analysis, the PDRC is assumed to remove its design capacity of 16.5 MW for a conservatism. When the pumps are turned off before 3,000 s, the current PDRC capacity is enough to cool down the sodium and it allows the original design capacity to be reduced up to 67 %. If the pumps were not turned off, the current capacity does not satisfy the safety limit as shown in Fig.4.

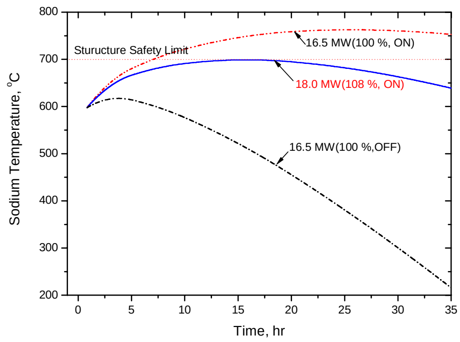

Figure 4 Pool sodium temperature (ULOHS, 2004)

4. Conclusion

The safety limits are always satisfied. The core inherent safety has also been guaranteed, because of the negative net reactivity feedback over the transients. The primary and intermediate sodium flows and the hot pool model seem to play dominant roles in a temperature transient, particularly, in a natural circulation during a ULOF. The current PDRC capacity is found to be sufficient enough for a long term cooling, and its capacity could be reduced up to 67 % of the current capacity.

All the calculation results are physically explainable and consistent. Although with the analysis methodology established in this study, a reliable conclusion can not be drawn until a complete set of the final design data is made in the future.

References

[1] Dohee Hahn et al., "Conceptual Design of Advanced Sodium-Cooled Fast Reactor KALIMER-600," Proceeding of ANS 2005 Annual Meeting, June 5-9, 2005, San Diego, California, U.S.A. (to be published).

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[3] W.P. Chang, "Safety Analysis for Key Design Features of KALIMER-600 Design Concept," KAERI/TR-2968/2005 (2005).