'00 춘계학술발표회 논문집
한국원자력학회

KALIMER 의 장기냉각 능력에 대한 평가

Assessment of KALMER for Long Term Cooling Capability

장원표, 권영민, 이용범, 석수동, 한도희

한국원자력연구소
대전광역시 유성우체국 사서함 105 호

요 약

KALIMER(Korea Advanced LIquid MEtal Reactor) 설계에서는 원자로의 열제거원이 상실되는 사고를 대비해 최종 열제거 방법중 하나로 피동열제거 특성을 가진 PSDRS (Passive Safety Decay Heat Removal System)를 이용한다. 이 계통은 자연순환을 이용해 격납용기 외부벽면을 냉각함으로써 원자로에서 발생된 열량을 제거한다. 본 연구는 개별적으로 개발된 PSDRS 모델을 계통분석 코드 SSC-K 에 연결한 후, 이를 이용해 KALIMER 계통의 열제거원 상실사고시 장기 열제거능력을 평가하여 개념설계의 안전해석 기준에 대한 개선점 제시를 목적으로 수행되었다.

본 연구에서는 노심 반응도 효과 분석을 통한 노심설계와 관련된 사항 외에도 운전원 조치시간, 그리고 사고해석 방법론 결정과 같은 안전해석기준 관련 내용에 대해서도 분석하였다. 또한 연구결과 PSDRS 모델이 추가되어 장기냉각 해석이 가능해진 SSC-K 의 전반적인 해석능력을 확인 할 수도 있었다. 향후에도 본 연구결과를 기본으로 KALIMER 설계의 안전성 증대를 위해 안전성 입증 및 개선 연구를 지속할 계획이다.

Abstract

The KALIMER design adopts PSDRS(**P**assive **S**afety **D**ecay **H**eat **R**emoval **S**ystem), which use a passive way to remove the decay heat, as an ultimate heat sink for the loss of heat sink accident. The system removes the heat generated in the reactor core by cooling the containment vessel wall through natural circulation. The present study is to assess the long term cooling capability of KALIMER, using the system analysis code SSC-K which is coupled with PSDRS model developed independently, for the purpose of proposal for improvements to the safety analysis criteria of the conceptual design.

The present paper presents the analysis results not only on the core reactivity feedback, but also on the operator action and the safety analysis methodologies concerned with the safety analysis criteria for KALIMER. As a result of the study, a qualification of SSC-K coupled with PSDRS on the analysis capability for the long term cooling has also been made. The study on the safety verification as well as design improvements will be continued for enhancement of the KALIMER safety.

I. Introduction

In the KALIMER design [1], the loss of heat sink accident is presumed to occur if IHTS (**I**ntermediate **H**eat **T**ransportation **S**ystem) is isolated to prevent the potential for the propagation of sodium-water chemical reaction in a sodium-to-water heat exchanger (Steam Generator), an IHTS pipe is ruptured, or the rupture disk bursts. [2] The accident is assumed to begin with a sudden loss of the normal heat sink by sudden stoppage of the IHTS flow. Natural circulation in IHTS is ignored so that this event is similar to complete loss of coolant in the IHTS, as might be true for the pipe rupture or the rupture disk burst. All heat generated in the core is, thus, retained in the primary vessel. PSDRS is designed to avoid such unlimited heat up of the primary system, which could lead to significant core damage and offsite release of radioactive material.

Since PSDRS is the only reliable system to stabilize heat up of the primary system under the accident, it has been designed to operate in a passive manner for reliability. PSDRS cools the outer surface of the containment through the circulation of atmospheric air. Relatively colder air enters the intake located on the top of the containment and flows downward through the annular gap between the outer divider wall and the inner concrete wall. The air then turns upward at the bottom and flows out through the annular gap between the containment outer wall and the inner divider as it gets hotter by cooling the containment wall. The air flow rate depends on the temperature difference between the inner and outer channels at center of the divider, form loss of the path, the pressure drop due to the orifice installed for the air flow control, and the wall friction inside the flow path. Therefore, insulation of the divider is very important to increase air flow rate. It is also noted that radiation heat transfer plays an important role, because helium gas is filled in the annulus gap between the inner and outer walls inside the primary vessel.

The reactor should be tripped due to high core outlet sodium temperature by the protection system, the reactor trip is, however, precluded in the present analysis. Such an unprotected accident is not design basis in the KALIMER design, but it has to be analyzed, because the design uncertainties and reliabilities are not ensured for the advanced design concept. The primary pumps are designed to operate at the rated conditions until tripped by the pump protection system, a safety grade system to open the pump breakers if the primary cold leg temperature exceeds the setpoint. The automatic control system and operator actions are of importance because the primary pumps contribute a major heat source to PSDRS for the long term cooling. In the present analysis, however, the primary pumps (four pumps) are also unprotected to find the effect of the pump operations during the accident.

Since PSDRS is the only safety-grade heat removal system for KALIMER [3], it is required to closely scrutinize its potential failure modes. Some degradation in the system performance is possible and some partial failures are also conceivable. The failure mode that seems most significant would be blockage of the air-flowing duct. However, the four independent air-flowing ducts are very large and less probable to be fully blocked, except via a massive earthquake or an extremity through act of sabotage.

Since PSDRS is such a crucial system, analyses are to be performed for two cases for comparison :

1) All airflow pathways in PSDRS are assumed to be fully blocked (Unavailable PSDRS case)
2) Assume 100% functioning of PSDRS (Available PSDRS case)

The main concerns in the analysis are to confirm inherent safety characteristic of the KALIMER core with the Plutonium fuel and to identify whether any abnormal behavior is found under the accident. The analysis is carried out using SSC-K [4], which is a modified version of SSC-L [5], developed in ANL for a loop-type Liquid Metal Reactor, and it is coupled with PSDRS model [6] independently developed for the KALIMER application.

II. Analysis Results

The basic assumptions considered in the present analysis are :

(1) The accident is assumed to start with a complete loss of the normal heat sink by sudden stoppage of the IHTS flow at time equals 0.0 sec. Thus, natural circulation in IHTS is ignored so that this event is similar to complete loss of coolant in the IHTS, as might be true for the pipe rupture or the rupture disk burst.

(2) A core protection system is not available. The primary pumps are neither automatically nor manually tripped during the whole transient.

In the present analysis the transient simulation is made for 40,000 sec (~ 11.1 hrs). The accident is initiated at 0.0 sec. Fig. 1 shows the core power and flow rate changes with time. As the accident occurs, the core heat generation drops rapidly due to the strong net negative reactivity of approximately 40 cents around 2,000 sec. On the other hand the core flow reduction is small and keeps almost 96 % of the initial flow because of the pump operation. The sodium and core radial expansion reactivities are identified as the major contributors to the early reactivity feedback as seen in Fig. 2. The net reactivity continues to decrease with ~ - 17.8 cents/hr. The reactivity after about 800 sec, however, does not affect the core power generation, because the decay heat is the primary core heat source after that time. In contrast with the case of the Uranium core, the Doppler reactivity is relatively less significant in the Plutonium core. In the present analysis the core protection systems are not assumed to be available, but the amount of reactivity introduced turns to be enough to shutdown the reactor. The core power after 800 sec tends to keep the decay heat level of about 25 MWt, which corresponds to approximately 6.3 % of the nominal power. As the power is not affected by the reactivity since after, it continues to decrease with the decay heat curve.

The loss of cooling in the IHX results in rapid increase of the cold pool temperature within a short time. As illustrated in Fig. 3, the cold pool temperature reaches almost the same value as that in the hot pool within about 500 sec and so does the core inlet temperature. Thereafter, the temperature increases very slowly with ~ 16 ^{o}C/hr, because of large heat capacity of the sodium in the pool as well as the PSDRS heat removal. About 120 ^{o}C is estimated for the subcooling margin at 40,000 sec as shown in Fig. 4. On the other hand, the increasing rate of the pool temperature without PSDRS shows ~ 32.2 ^{o}C/hr which is faster than that of the case with PSDRS by almost two times. The cladding temperature exceeds the safety limit (977 ^{o}K) mainly due to the pump heat rather than the decay heat. The calculated cladding temperature is about 1075 ^{o}K and the core heat generation is about 2.9 MWt (0.7 % of the nominal power) at 40,000 sec. This decay heat level is well below the PSDRS heat removal (Fig. 5) but the pool temperatures rise without stabilizing. That means 2.8 MW of the heat generation from the total 4 pumps becomes an important heat source as the decay heat gets smaller and the cladding safety is not ensured without elimination of the pump operation. Although PSDRS effect is clear in Fig. 6, the maximum sodium coolant, cladding, and fuel temperatures in the core rise linearly following an abrupt change at the early time. The rapid decrease of the fuel center temperature must attribute to the core power reduction resulting from the reactivity feedback. As a characteristic of the metal fuel, the temperatures of the core coolant, cladding, and fuel behave similarly. The sodium coolant does not show any abnormal response and the ratio of the core flow rate to the core power is high enough so that a peak cladding temperature in the core is not likely to occur.

The peak behaviors shown in Fig. 3 through 6 around 700 sec, attribute to the reactivity feedback. As the core temperature goes up, the net reactivity also increases due to the positive sodium reactivity while the negative reactivity for the radial expansion drops more rapidly around 700 sec. (Fig. 2) As the result, the increasing net reactivity turns down and, thus, the core coolant temperature decreases for a short time, which leads to the sodium reactivity change. After then the reactivity due to the core radial expansion changes smoothly and the core coolant temperature also rises slowly. The slope change around 5,900 sec in Fig. 5 indicates that the hot pool sodium spills out into the cold pool so that the sodium level in the cold pool is increased and the PSDRS heat removal is enhanced. The amount of the over-flow is represented in Fig. 7 and it is slightly over than 50 % of the core flow.

3. Discussion and Conclusion

In the present simulation consistent results are obtained and could be explained on the physical basis. The effect of PSDRS and its success of coupling with SSC-K are clearly demonstrated. Therefore, overall prediction by SSC-K seems to be reasonable. The inherent safety in the KALIMER design induced from the strong negative reactivity under ULOHS (Unprotected Loss Of Heat Sink) accident is also addressed through the present analysis. It is noticed that the net negative reactivity mainly results from the competition between those for the sodium and radial expansion, while the sodium reactivity is not so sensitive to the net reactivity in the Uranium core design. [7]

Regardless of those desirable effects, some problems for safety are still remained. The first thing to be mentioned is the assumption for the pump operation during the accident. According to the safety criteria developed for the KALIMER design, no operator action is allowed for 72 hrs after accidents. As described in the previous section, the pump power plays an important source of the heat generation in the primary vessel and it limits energy balance for long term cooling. There seems to be two choices. One is to give the credit to the pump breaker to satisfy the 72 hr goal without an operator intervention. In this case analysis of ULOHS alone has no meaning and it should be included in the category for the analysis of ULOF/ULOHS (Unprotected Loss Of Flow / Unprotected Loss Of Heat Sink). Another option is to enhance the PSDRS capability by its design change to the forced circulation of air. The reliability for the ultimate heat sink is much concerned to replace the passive component with safety-grade active component. Another point is that the early sodium reactivity insertion of ~ 130 cents is considered to be too much comparing with that for the Uranium core (a few cents). The design effort for the core should be followed to reduce this value and then the time for the pool sodium temperature taken to reach the safety limit will be much prolonged. All these safety concerns will be examined more carefully as the design develops.

Further studies, therefore, should be made to define the safety analysis criteria related with DBE (Design Basis Events) methodologies as well as to improve the design. Both analytical and experimental efforts, however, must be continued as well in such areas as PSDRS effect and validation of SSC-K for the complete safety analysis of the KALIMER design.

Acknowledgement

This work was performed under the Long-term Nuclear R & D Programs sponsored by the Korea Ministry of Science and Technology.

References

[1] Chang Kyu Park et al., ‘KALIMER Design Concept Report’, KAERI/TR-888/97

[2] ‘Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor,’ NUREG-1368

[3] Y.M. Kwon, et al., ‘Preliminary Safety Design Analysis of KALIMER,’ KAERI/TR-1270/99

[3] K.D. Kim et al., ‘Development of Pool Thermal-hydraulic Analysis Model for SSC-K,’ KAERI/TR-1266/99

[5] J.G. Guppy, et al., ‘Super System Code (SSC, Rev. 0) An Advanced Thermal-hydraulic Simulation Code for Transient in LMFBRs,’ NUREG/CR-3169

[6] W.P. Chang, et. al., ‘Development of the PSDRS Model for the KALIMER System Analysis code, SSC-K’, KNS Fall Meeting, 1998

[7] W.P. Chang, ‘ULOHS 분석’, LMR/SA131-AR-03/99, KAERI Internal Report

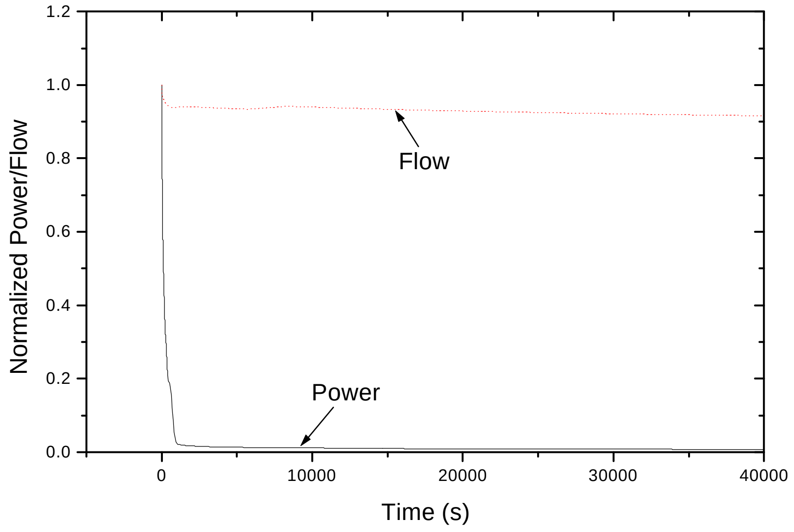

Fig. 1　Core Power and Flow

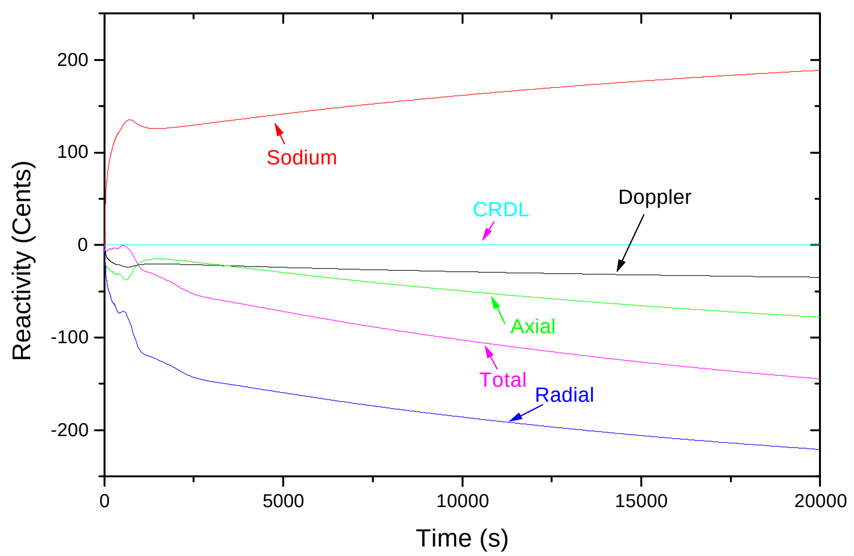

Fig. 2　Reactivity

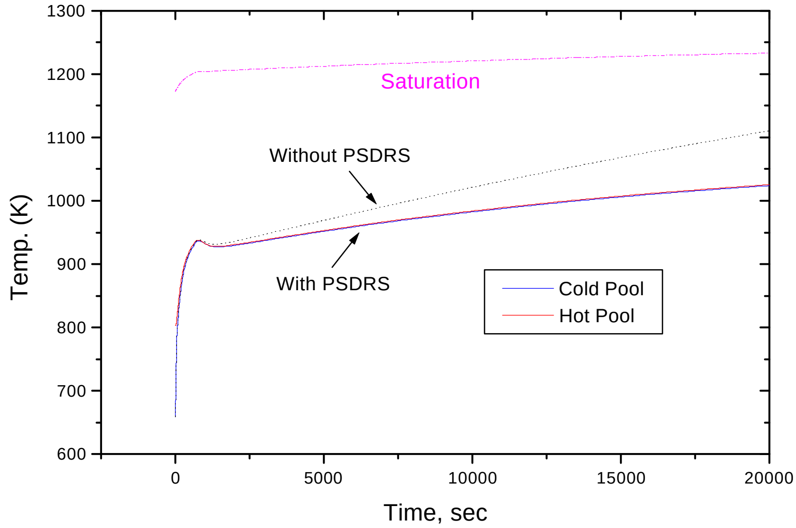

Fig. 3 Pool Temperatures (w/ , w/o PSDRS)

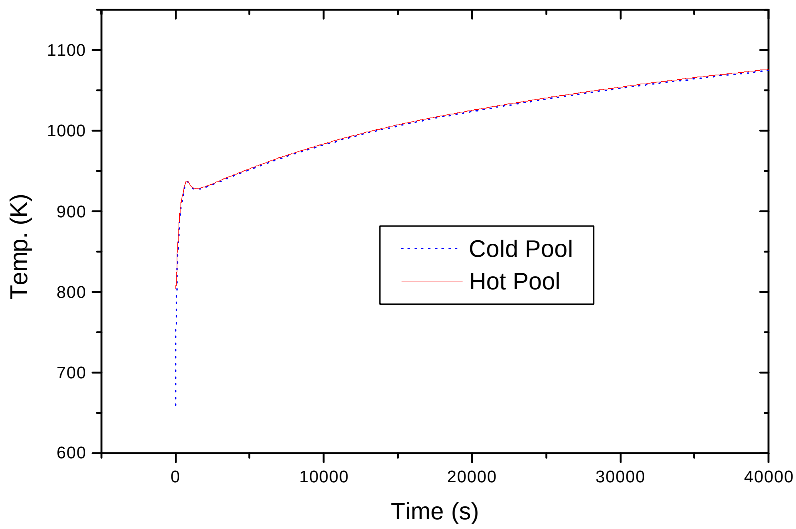

Fig. 4 Temperatures for Hot and Cold Pools

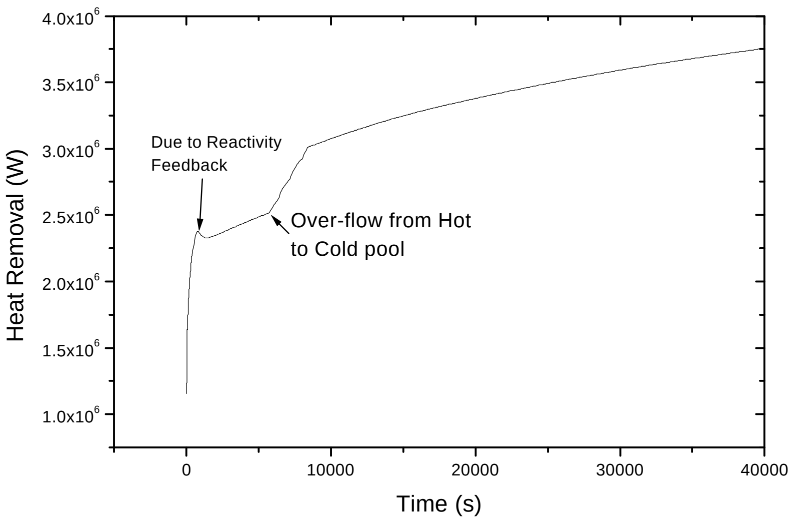

Fig. 5   PSDRS Heat Removal

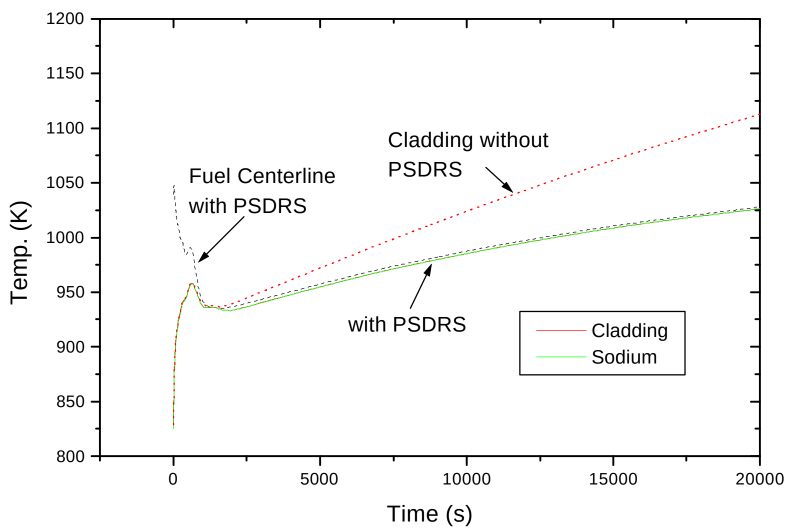

Fig. 6   Maximum Fuel, Cladding, and Sodium Coolant

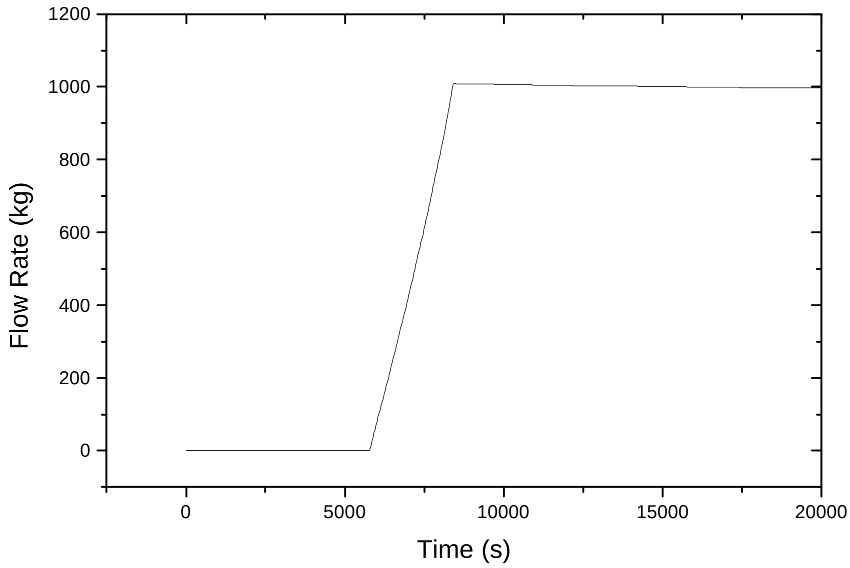

Fig. 7 Over-flow from Hot Pool to Cold Pool