# Accident Analysis of Steam Line Break with Common Cause Failure in Digital Instrumentation and Control Systems

Ki Moon Park*, Jong Cheol Park, Chan Eok Park, Gyu Cheon Lee, Jae Young Huh

*Safety Analysis Group, KEPCO-E&C, 989-111, Daedeokdaero, Yuseong-gu, Daejeon, 34057*

**Corresponding author: powermoon@kepco-enc.com*

## Introduction

Digital instrumentation and control (DI&C) systems can be vulnerable to common-cause failure (CCF) caused by software errors or software developed logic, which could defeat the redundancy achieved by hardware architecture [1]. The purpose of design basis event (DBE) with a concurrent CCF in the DI&C systems (hereafter DBE with CCF) is performed for support analysis to demonstrate diversity and defense-in-depth (D3).

Steam line break (SLB) outside containment with a concurrent CCF (hereafter SLB with CCF) is quantitatively evaluated with respect to offsite doses limit according to the acceptance criteria [1]. Since Fukushima accident, regulatory body has begun to pay more attention to beyond design basis events (BDBEs) as important events related to safety enhancement and even though the frequency of occurrence is very low, the relevant guidelines are being strengthened. Under these circumstance, the new safety issues for SLB with CCF are raised by Korean regulatory body during the licensing of constructing nuclear power plants (NPPs), and it is identified that when the evaluation of SLB with CCF considering the new safety issues is performed, it could be a challenge to acceptance criteria.

In this paper, the effect of the new safety issues on SLB with CCF is evaluated. In addition, countermeasure is derived for mitigating its consequences and the results of SLB with CCF applying derived countermeasure are also shown.

## Methods and Results

SLB with CCF is classified as a BDBE, and this analysis can use best-estimate methods (realistic assumptions). The thermal hydraulic response of the nuclear steam supply system (NSSS) was simulated using the CESEC-III computer program [2]. Fuel temperature was calculated using the STRIKIN-II computer program [3], which is not currently used due to the code limitation.

### 2.1 Acceptance Criteria

The SLB is categorized as postulated accident. The acceptance criteria for the postulated accident with CCF are presented on the SRP BTP 7-19 [1], which are as follows:

- For each postulated accident in the design basis occurring in conjunction with each single postulated common cause failure, the plant response calculated using best-estimate (realistic assumptions) analyses should not result in radiation release exceeding 10 CFR 100 guideline values, violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment.

The SLB is defined as a pipe break in the main steam system of outside containment, which results in excessive reactor coolant system (RCS) cooldown and cause the primary pressure to decrease. Therefore, the integrity of the primary coolant pressure boundary and the integrity of the containment are not significant for SLB with CCF. With respect to the offsite doses limit, the fuel melting is conservatively chosen as an indicator to satisfy the dose limit since the occurrence of fuel melting would give a significant effect on core coolability as well as the offsite dose.

### 2.2 Assumptions

The major assumptions of this best estimate analysis are as follows:

1) Normal full power operating conditions are used in the analysis.
2) Additional single failures are not assumed.
3) Non-safety NSSS control systems are normally assumed to be operable.
4) Initial conditions of operating parameters are assumed at their normal operating condition and those of control systems are assumed to be operated with their nominal values.
5) The diverse protection system (DPS) of reactor trip functions and engineered safety features actuation functions are assumed not to be affected by the CCF.
6) The offsite power is assumed to be available, hence Reactor Coolant Pumps (RCPs) are assumed to be normally operating.

### 2.3 The new safety issues

The Korean regulatory body requested additional information for the long term SLB with very high core power by considering as follows;

1) Xenon depletion
2) The physics data such as axial power distribution (APD) and radial peaking factor (Fr) after transient occurrence
3) Reload fuel data
4) Thermal conductivity degradation (TCD)

### 2.4 The effect of the new safety issues on SLB with CCF

#### 2.4.1 The effect of Xenon depletion

The large energy extraction caused by the steam line break reduces the RCS temperature. The resulting cooldown causes a rapid increase in core power, and core power is maintained for a long time without reactor protection system (RPS) reactor trip. For this case, the positive reactivity due to Xenon depletion is added and the core power will be further increased.
As shown in Figures 1, core power of re-analysis is a bit higher than that of the previous result, because the Xenon depletion causes positive reactivity insertion into the reactor core.

#### 2.4.2 The effect of physics data (APD and Fr) after transient

The 3-D Peaking Factor (Fq) increases since the APD is shifted downward due to lower core inlet temperature and much higher core power for the event.
Because of Fq change, fuel centerline temperature rise is very significant. As shown in Figure 2, fuel temperature is much higher than the previous result since the core condition with high power and seriously bottom skewed APD are maintained for a long time. However, the calculated maximum fuel centerline temperature is identified as being below fuel melting temperature.

#### 2.4.3 The effect of reload fuel data and TCD

Important nuclear physics parameters used in SLB with CCF are moderator cooldown reactivity (MCR), fuel temperature coefficient (FTC) and kinetic parameter. These nuclear physics parameters of reload fuel are more conservative than these of the initial fuel data, for overcooling event.
Core power will increase further when considering reload fuel data, and fuel centerline temperature will also increase even more if TCD is considered.

### 2.5 Countermeasure

Fuel temperature is already close to melting point by only two (Xenon depletion and physics data after transient) of the aforementioned four issues, and it is expected that the result of SLB with CCF could be challenged to the fuel melting and the offsite dose results may not meet the acceptance criteria if additionally applying the remaining two issues (reload fuel data and TCD).
The decisive reason for the worsening of the SLB with CCF results with the new safety issues is that very high core power lasts for a long time due to the excess cooldown from the SLB without reactor trip. So, the most immediate and effective countermeasure to mitigate the SLB with CCF results is early reactor trip after event initiation.
The actions for early reactor trip to prevent expansion of the worsening the SLB with CCF results are as follows;

1) Credit earlier reactor trip by operator
2) DPS design change

No operator action until 30 minutes after event initiation is conservatively assumed in the previous analysis of SLB with CCF. However, the reasonable operator action before 30 minutes after the event can be credited in the evaluation of SLB with CCF. Therefore, the manual operator action for actuation of reactor trip could be considered in the analysis as immediate actions to mitigate the SLB with CCF results.
Of the two actions above, the DPS design change is more effective countermeasure to resolve the fundamental problem and improve safety of NPPs since the early operator action (reactor trip) is a considerable burden on the operator.
In this regard, the DPS design change is promoted and a new low steam generator pressure (LSGP) reactor trip function is implemented in the DPS for all domestic NPP for designs using digital protection systems.
Various sensitivity analysis results and range of RPS LSGP reactor trip are considered in the determination of the DPS LSGP reactor trip setpoint.

### 2.6 The results of SLB with CCF with the DPS LSGP reactor trip

Thermal hydraulic behaviors are shown in Figure 3 through Figure 5. The SLB will lead to a depressurization of the steam generator due to the loss of secondary coolant inventory, which results in a reactor trip on LSGP by the DPS at about 10 seconds after event initiation. The resulting cooldown causes a rapid increase in core power which is calculated to peak at approximately 140%. The fuel centerline temperatures follow the same trend as the power, r remains below the fuel melting temperature and the calculated offsite radiological doses are well within SRP BTP 7-19 [1].

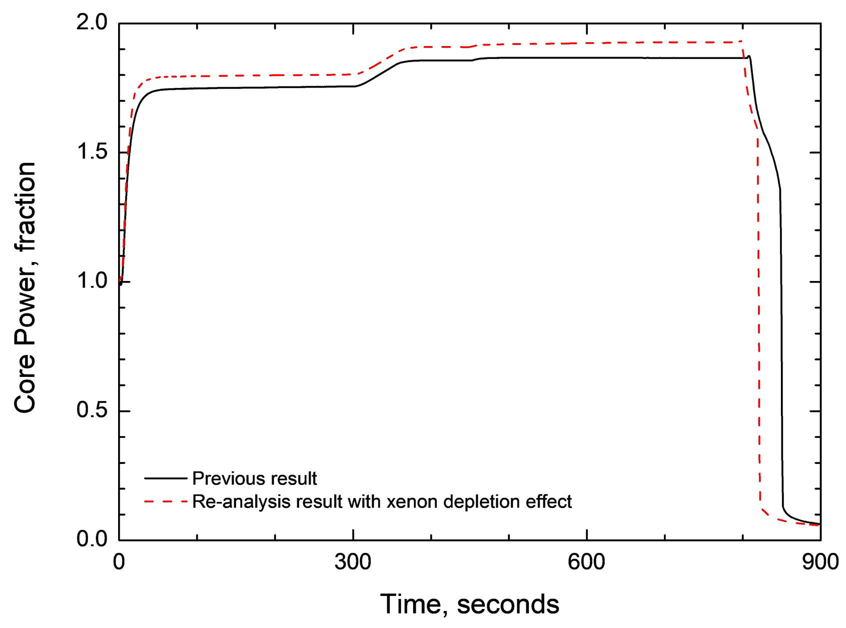


Fig. 1. Core Power (SLB with CCF)

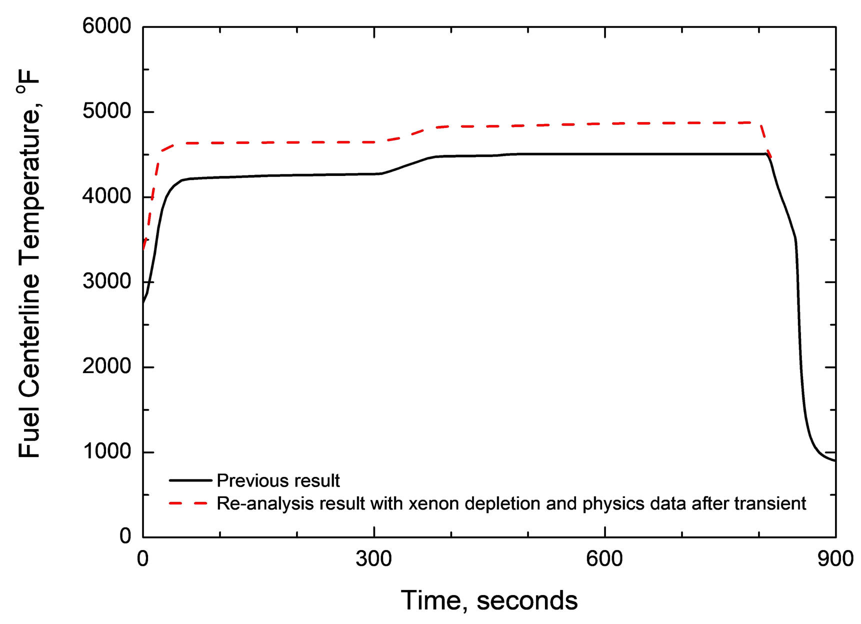


Fig. 2. Fuel Centerline Temperature (SLB with CCF)

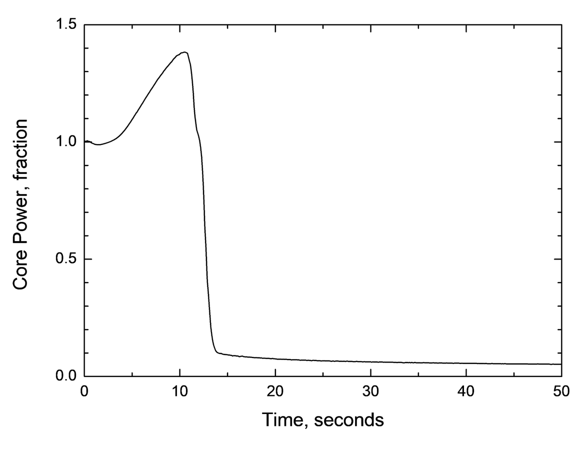


Fig. 3. Core Power (SLB with CCF)

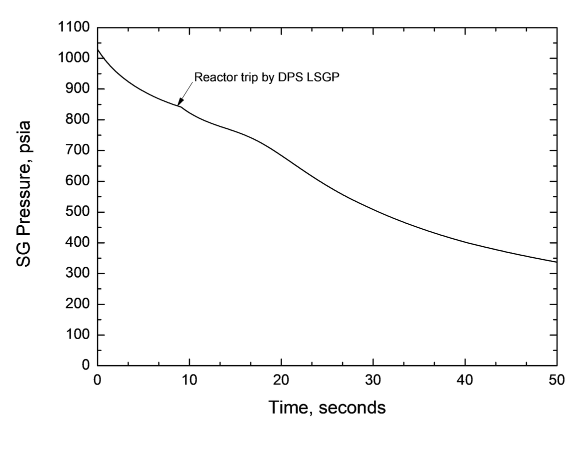


Fig. 4. SG Pressure (SLB with CCF)

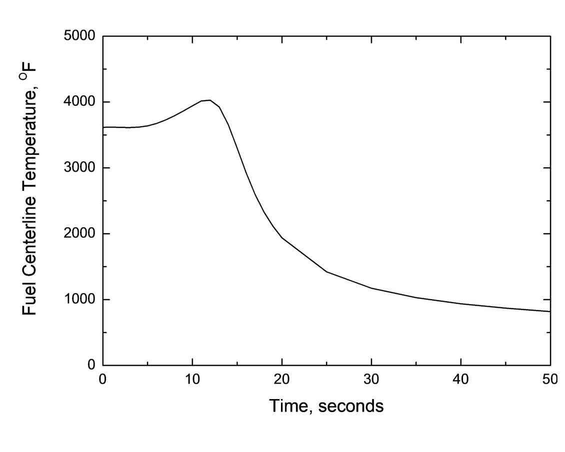


Fig. 5. Fuel Centerline Temperature (SLB with CCF)

## Conclusions

SLB with CCF has been re-evaluated with new safety issues raised by Korean regulatory body and it is expected that the results of SLB with CCF could be challenged to the fuel melting and the offsite dose results may not meet the criteria.
Therefore, the DPS design change is promoted and it is identified that LSGP reactor trip newly added in the DPS as a countermeasure can successfully mitigate its consequences.

## REFERENCES

[1] "Guidance for evaluation of diversity and defense-in-depth in digital computer-based instrumentation and control systems review responsibilities," NUREG-0800 BTP 7-19 R7, Aug. 2016.
[2] "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," CENPD-107, April 1974.
[3] CENPD-135, STRIKIN-II, "A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1974.