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# A Study on Effects of Initial Conditions to the Fuel Integrity Analysis of Steam Generator Tube Rupture Accident

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## 1. Introduction

Steam generator tube rupture (SGTR) accident is a penetration of the barrier between reactor coolant system (RCS) and secondary system that results from the failure of a steam generator U-tube found in Fig. 1. During SGTR accident, RCS pressure continuously decreases while the core power, core flow rate and core average temperature almost do not change until reactor trip occurs. As a result, the departure from nucleate boiling ratio (DNBR) also continuously decreases, thus eroding the thermal margin to DNB. To identify no damage of the fuel cladding due to the onset of film boiling, it must be analyzed whether the heat flux is maintained below the critical heat flux or not [1]. For this purpose, it is used to define the specified acceptable fuel design limit (SAFDL) of DNBR and evaluate that the DNBR stays above the SAFDL.

Evaluating the DNBR in SGTR accident, several initial conditions are varied as that would challenge to the safety limit, or SAFDL. However, the effects of some initial conditions are complicate and difficult to be intuitively identified. Thereby, in this study, the effect of initial conditions on DNBR is analyzed.

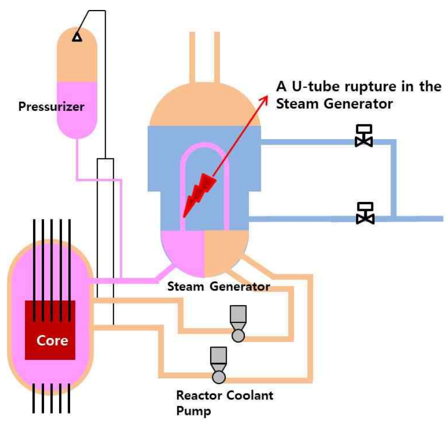


Fig. 1 Schematic Diagram of Steam Generator Tube Rupture Accident

## 2. DNBR Analysis for SGTR Accident

Analyzing SGTR accident, a loss of offsite power (LOOP) concurrent turbine trip is assumed for the conservative result with respect to the DNBR. Due to the coast down of reactor coolant pump (RCP) flow following reactor trip, the core flow rate decreases faster than the core heat flux making the DNBR decrease very rapidly during a brief period of time.

The initial conditions that mainly affect the minimum DNBR during SGTR accident are the initial RCS flowrate, the initial core power, the initial core inlet temperature, and the initial pressurizer pressure. Among them, the initial RCS flowrate and core power can be intuitively determined as minimum and maximum, respectively. Those conditions increase the core temperature abating the thermal margin to DNB. The maximum initial core inlet temperature and the minimum initial pressurizer pressure can make the lowest initial DNBR. However, in SGTR accident, the minimum DNBR does not appear at the initiation of the accident but typically appears far after the accident initiation. Also, the RCS pressure continuously varies during SGTR transient. From these reasons, conservative initial conditions for the core inlet temperature and the pressurizer pressure are difficult to be intuitively chosen. Also, based on the analysis experiences, it has ever been known that the minimum initial core inlet temperature and the maximum initial pressurizer pressure can make the lowest minimum DNBR in SGTR accident.

Therefore, the effects of these two initial conditions are studied in the following sections.

## 3. Study on Effect of Initial Conditions

In this work, the advanced power reactor (APR) 1400 is chosen as the reference plant. The thermal hydraulic response of the nuclear steam supply system (NSSS) during SGTR accident is simulated using the CESEC-III computer program [2]. The DNBR in the reactor core is calculated using the CETOP computer program with the KCE-1 critical heat flux correlation [3]. The analysis cases used in this study are as shown in Table I.

Table I: Initial Conditions of the Test Cases

| Case | Core Inlet[1)] Temperature | Pressurizer[2)] Pressure |
|---|---|---|
| 1 | Minimum | Minimum |
| 2 | Maximum | Minimum |
| 3 | Minimum | Maximum |
| 4 | Maximum | Maximum |

[1)] Maximum = 295℃ Minimum = 287.78℃
[2)] Maximum = 152.92 kg/cm$^2$ Minimum = 163.46 kg/cm$^2$

### *3.1 Initial Core Inlet Temperature*

The transients of the core inlet and outlet temperature during SGTR accident are shown in Fig. 2. From the results, both the core inlet and outlet temperatures are regarded as almost constant before reactor trip since the heat addition by the fuel is stable until reactor trip by hot leg saturation.

After SGTR occurrence, the RCS pressure continuously decreases, hence the relevant saturation temperature. That is, the lower hot leg temperature causes the later reactor trip by the hot leg saturation trip function. Because the minimum initial core inlet temperature may produce the minimum initial hot leg temperature and the RCS temperatures are almost maintained at the initial values until reactor trip during SGTR accident, the assumption of the minimum initial core inlet temperature most postpones reactor trip by the hot leg saturation trip function as shown in Fig. 3.

### *3.2 Initial Pressurizer Pressure*

The RCS pressure transients during SGTR accident are shown in Fig. 4. As above mentioned, after SGTR initiation, the RCS pressure relevant saturation temperature continuously decreases. Therefore, assuming the maximum initial pressurizer pressure consequently most puts off reactor trip by the hot leg saturation trip function as shown in Fig. 5.

### *3.3 Impact of Reactor Trip Delay*

After SGTR occurrence, the DNBR continuously decreases with almost constant rate until reactor trip since the RCS pressure continuously decreases while the reactor power is constantly maintained. Therefore, the reactor trip is delayed more, the DNBR decreases more.

Also, in SGTR accident, the reactor trip delay affects the RCS inventory at the time point of reactor trip.

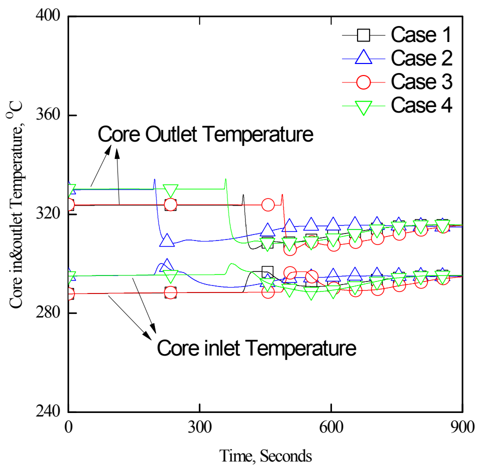


Fig. 2 Core Inlet & Outlet Temperature Transients

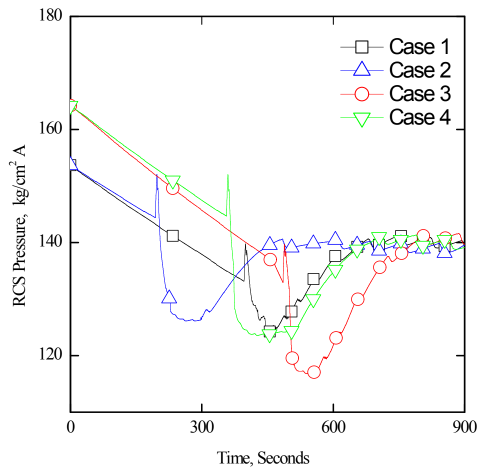


Fig. 4 RCS Pressure Transients

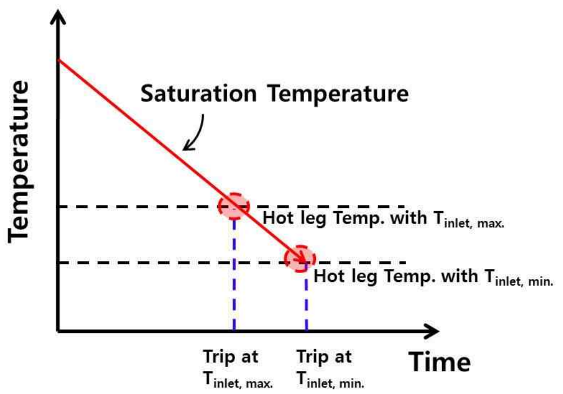


Fig. 3 Comparison of Reactor Trip Time Points Depend on the Initial Core Inlet Temperatures

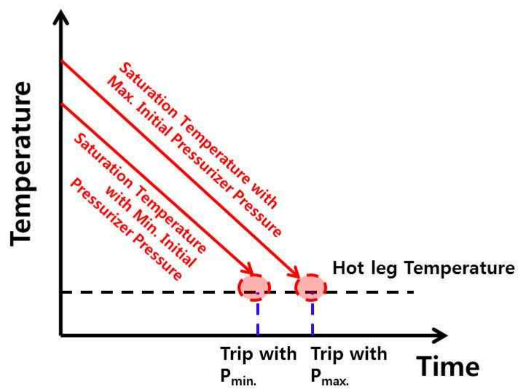


Fig. 5 Comparison of Reactor Trip Time Points Depend on the Initial Pressurizer Pressures

That is, later reactor trip in SGTR makes smaller RCS inventory at the reactor trip since the RCS coolant continuously leaks through the ruptured U-tube.

When the reactor trip occurs with a LOOP, the DNBR abruptly decreases right after the reactor trip then increase. And it is identified that the smaller RCS inventory at the time point of reactor trip tends to make this DNBR undershoot, under a LOOP condition, larger as shown in Fig. 6 and Fig. 7.

Consequently, it is known that the maximum initial pressurizer pressure and the minimum initial core inlet temperature, shown as case 3 in Table II, can make the lowest minimum DNBR in SGTR accident through reactor trip delay.

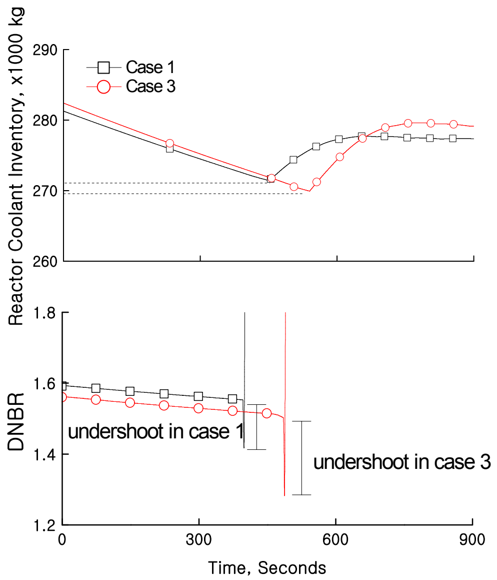


Fig. 6 RCS Inventory and the Corresponding DNBR Results Using the Minimum Core Inlet Temperature

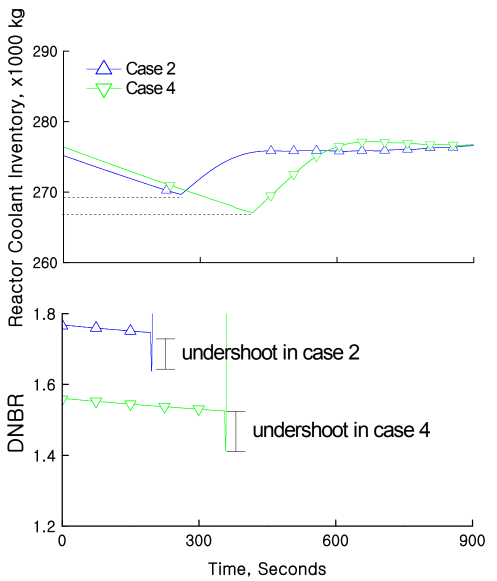


Fig. 7 RCS Inventory and the Corresponding DNBR Results Using the Maximum Core Inlet Temperature

Table II: Minimum DNBR Results from the Test Cases

| Case | MDNBR | Remark |
|---|---|---|
| 1 | 1.4171 | - |
| 2 | 1.6382 | - |
| 3 | 1.2818 | Limiting Case |
| 4 | 1.4124 | - |

## 4. Conclusions

The effects of initial conditions to the DNBR are studied for SGTR accident in this work. The initial conditions mainly affect the minimum DNBR during SGTR accident are the initial core power, the initial RCS flow rate, the initial core inlet temperature and the initial pressurizer pressure. The conservative initial conditions for the core power and the RCS flow rate can be determined easily by qualitative evaluation. The others such as the initial core inlet temperature and the initial pressurizer pressure are complicate and difficult to be determined.

From this study, it is identified that the maximum initial pressurizer pressure and the minimum initial core inlet temperature on SGTR accident most postpones the reactor trip resulting in smaller RCS inventory at the reactor trip time point. And the smaller RCS inventory at the time point of the reactor trip tends to make the DNBR undershoot larger which is a dominant factor to determine the minimum DNBR under a LOOP condition.

## REFERENCES

[1] Introduction to Nuclear Engineering, 3rd Edition, John R. Lamarsh, Anthony J. Baratta, 2001

[2] CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System, 1981.

[3] Software Verification and Validation Report for CETOP Ver. 1 Mod4 kce1, KEPCO NF, 2004.