

Preliminary Analysis of Severe Accidents in a 1400 MWe-Class Generic PWR Using CINEMA

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1. Introduction

The analysis of phenomena occurring during severe accidents in nuclear power plants is essential for understanding accident progression and evaluating accident management strategies. In Korea, KHNP, KAERI, KEPSCO E&C, and FNC formed a consortium to develop CINEMA (Code for INtegrated severe accident Evaluation and Management), an integrated severe accident analysis code for light-water reactors[1]. Validation studies have been conducted against a range of experimental and plant data [1,2], and CINEMA has been applied to severe accident analyses for various reactor designs, including the OPR1000 [3] and i-SMR [4].

In this study, severe accident analyses were performed using CINEMA for a large-scale nuclear power plant more representative of commercial reactors than those previously analyzed. For this purpose, an input model for a 1400 MWe-class Generic PWR was developed, and preliminary severe accident analyses were conducted.

2. Scenario Selection

Preliminary analyses were performed for two representative accident sequences corresponding to low- and high-pressure accident conditions.

For the low-pressure accident sequence, a large-break loss-of-coolant accident (LLOCA) initiated by a 9-inch cold-leg break was selected as the representative scenario. The LOCA was assumed to occur at 0 s. Except for the Safety Injection Tanks (SITs), which operate passively without requiring a separate actuation signal or operator action, all other safety systems were assumed to be unavailable. This represents a highly conservative severe accident scenario. The accident progression was analyzed for the first 50,000 s.

For the high-pressure accident sequence, a station blackout (SBO), involving a complete loss of AC power, was selected as the representative scenario. The SBO was assumed to occur at 0 s, resulting in reactor trip, shutdown of the reactor coolant pumps (RCPs), and loss of secondary-side feedwater. As in the LLOCA scenario, all safety systems except the SITs were assumed to be unavailable, representing a highly conservative condition. However, maintaining the reactor coolant system (RCS) at high pressure throughout the accident

was considered excessively severe. Therefore, depressurization was assumed to be initiated 30 min after entry into the Severe Accident Management Guideline (SAMG) conditions by manually opening the pressure control valve located at the top of the pressurizer. The SBO sequence was also analyzed for 50,000 s.

Table I presents the differences between the specified design parameters of the generic PWR and the corresponding steady-state results calculated using CINEMA. Several thermal-hydraulic parameters, including reactor power, flow rates, pressures, temperatures, were selected for comparison under normal operating conditions. The results showed that the calculated initial steady-state conditions agreed well with the target values, with a maximum deviation of less than 3%.

Table I: Deviation between design values and CINEMA results under normal operating condition

Parameter	Deviation between design value and CINEMA results
Reactor power	0.00%
Core flow rate	-0.15%
Hot leg flow rate	0.01%
Cold leg flow rate	0.02%
Pressurizer pressure	0.00%
Core inlet temperature	0.51%
Core outlet temperature	0.44%
Steam generator pressure	0.43%
Steam generator level	2.81%

3. Results and Discussion

3.1 LLOCA Scenario

Figure 1 shows the RCS pressure during the LLOCA sequence. Following coolant discharge through the break and the resulting decrease in RCS pressure, the SITs were actuated at approximately 170 s. Because no core cooling systems other than the SITs were available, the remaining RCS coolant rapidly boiled off, resulting in an increase in core temperature. Hydrogen generation from the cladding began at approximately 1,360 s, and the SAMG entry criterion, defined by a core-exit gas temperature of 923 K, was reached at approximately 1,420 s.

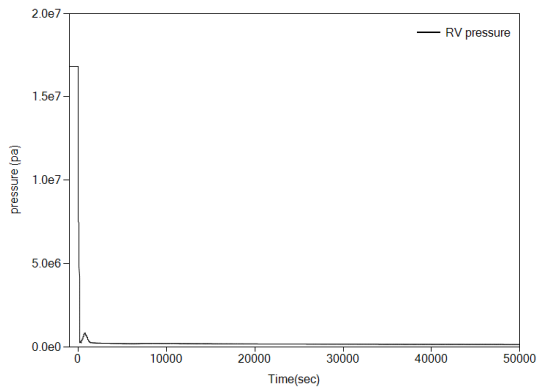


Fig. 1. RCS pressure for LLOCA scenario

As the core temperature continued to increase, core melting progressed. The first failure of the core support plate (CSP) occurred at 4,928 s, allowing molten material to relocate into the lower plenum (LP). Subsequently, failure of the reactor vessel (RV) lower-head penetration occurred at approximately 6,420 s.

Figure 2 shows the amount of hydrogen generated during the LLOCA sequence. The final hydrogen generation was calculated to be 121.2 kg. The relatively small amount of hydrogen was attributed to the rapid melting of zirconium (Zr) as the core temperature increased. Consequently, a significant fraction of the Zr melted before sufficient oxidation through reaction with steam could occur.

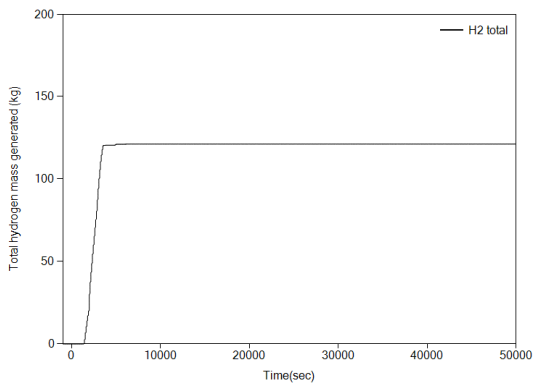


Fig. 2. Total hydrogen generation at the RV during LLOCA scenario

3.2 SBO Scenario

Figure 3 shows the RCS pressure during the SBO sequence. Following the reactor trip, RCP shutdown, and loss of secondary-side feedwater, evaporation of the RCS coolant caused the system pressure to increase. The pressure was subsequently maintained within a certain range by chattering of the pressurizer pilot-operated safety relief valve (POS RV).

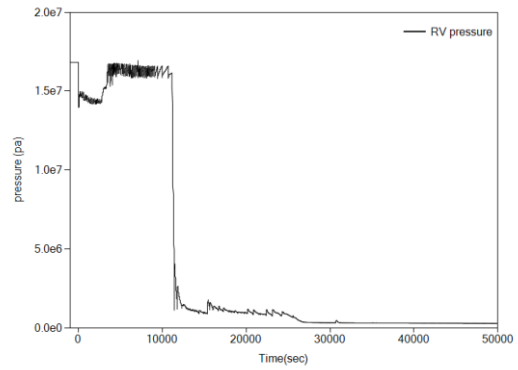


Fig. 3. RCS pressure for SBO scenario

As the coolant inventory was depleted, the SAMG entry criterion was reached at 9,440 s, and hydrogen generation from the cladding was observed at 10,644 s. At 30 min after SAMG entry, a depressurization action was initiated by manually opening the POSRV. As a result of the depressurization, partial core cooling by the SITs occurred at approximately 11,450 s, thereby delaying the subsequent accident progression. Nevertheless, CSP failure occurred at 30,726 s, followed by failure of the RV lower-head penetration at 34,713 s.

Figure 4 shows the amount of hydrogen generated during the SBO sequence. The final hydrogen generation was calculated to be 592.1 kg. Compared with the LLOCA sequence, the SBO progressed more slowly, providing sufficient time for Zr in the core to undergo oxidation. This is considered to be the primary reason for the substantially greater amount of hydrogen generated during the SBO sequence.

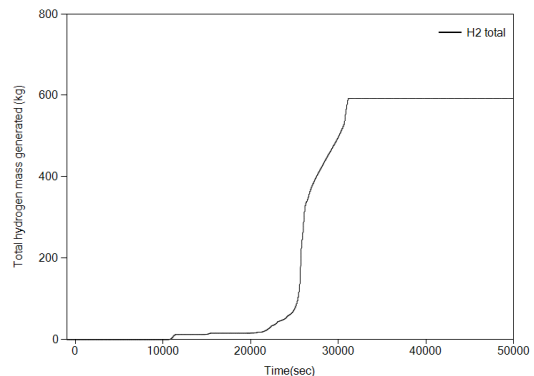


Fig. 4. Total hydrogen generation at the RV during SBO scenario

4. Conclusions

A severe accident analysis input model for a 1400 MWe-class Generic PWR was developed using CINEMA, and preliminary analyses were performed for two major severe accident sequences, namely LLOCA and SBO. The results showed that, in the absence of

additional safety systems, both the LLOCA and SBO sequences progressed to severe core damage, including core melting and eventual reactor pressure vessel (RPV) failure. The analyses also provided an overview of the fundamental accident progression characteristics under representative low- and high-pressure accident conditions.

The results of these preliminary base-case analyses will be used to further improve the completeness of the Generic PWR input model and to provide a basis for evaluating additional accident scenarios involving the operation of various safety systems.

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