

## Severe accident analysis of C-SGTR and S-SGTR at APR1400 using CINEMA code

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### 1. Introduction

The CINEMA code, an integral severe accident analysis code for light water reactors (LWRs), is currently applied in a range of domestic nuclear safety fields in Korea, including i-SMR licensing and safety assessments of operating nuclear power plants. This study presents a severe accident analysis of two representative steam generator tube rupture (SGTR) scenarios—Consequential SGTR (C-SGTR) and Spontaneous SGTR (S-SGTR) [1,2]—performed using the CINEMA code.

### 2. Methods and Results

The analysis was conducted for the APR1400 reactor design. The accident conditions, sequence of events, and key results for each case are summarized below.

Table I: Specification of C-SGTR and S-SGTR

| ESF/MODEL      | C-SGTR                 | S-SGTR               |
|----------------|------------------------|----------------------|
| SIT            | O                      | O                    |
| SIP            | X                      | X                    |
| HL-CREEP       | O                      | X                    |
| SG-UTUBE-CREEP | O                      | X                    |
| POSRV          | O                      | O                    |
| AFW            | X                      | X                    |
| RCP-leakage    | O                      | X                    |
| TRIP           | C-SGTR                 | S-SGTR               |
| Initial event  | Loss of class-IV power | Rupture of SG U-tube |
| Rx trip        | Time =0.0s             | MSIS                 |
| MFW trip       | Time =0.0s             | MSIS                 |
| MSIV trip      | Time =0.0s             | MSIS                 |
| TBN trip       | Time =0.0s             | MSIS                 |
| RCP trip       | Time =0.0s             | Cavitation trip      |

\*MSIS : Main Steam Isolation Signal

#### 2.1 C-SGTR

C-SGTR refers to an accident in which the high-pressure boundary of the primary system fails at an SG U-tube following an initiating loss of offsite power. As in a typical station blackout (SBO) sequence, the accident progresses through the following stages in order: a decrease and eventual dry-out of the secondary-side SG water level, depletion and pressurization of the primary coolant, maintenance of primary pressure at the

Pilot-Operated Safety Relief Valve (POSRV) setpoint, and finally reactor vessel failure.

Failure of the SG U-tube was modeled using a creep rupture model, and the rupture occurs when the U-tube heat structure temperature reaches around 1200°C. At this point, both steam generators have already dry-out, so fission products transported from the core and primary system are not scrubbed by SG secondary water and are released directly to the auxiliary building through the Main Steam Safety Valves (MSSVs).

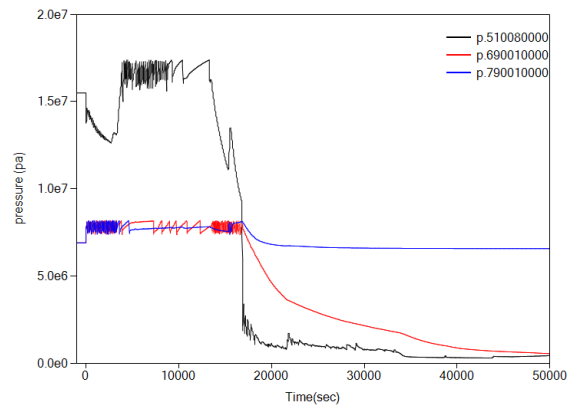


Fig. 1. Pressure of C-SGTR (c510-PZR, c690-Ruptured SG, c790-Intact SG)

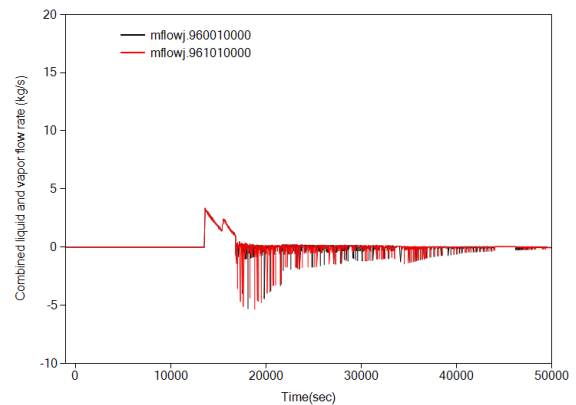


Fig. 2. Mass flow rate at U-tube break for C-SGTR (Double ended guillotine break)

## 2.2 S-SGTR

S-SGTR is initiated by a double-ended guillotine break of a single SG tube. From the onset of the accident, high-pressure primary coolant flows into the secondary system, and the primary pressure is subsequently maintained at the secondary-side MSSV setpoint. When the coolant in SG is dryout, pressure of primary loop maintains at setpoint of POSRV.

Interesting point is that (i) modeling of SG u-tube break strongly impacts mass flow rate between primary and secondary loop such as offtake option and (ii) pressure of ruptured SG after the vessel failure, its depressurization rate is limited by the choking at the u-tube break because of its small size (area of SG single u-tube break = 0.0002248 m<sup>2</sup> by I.D. of SG U-tube = 16.918mm vs. area of unit POSRV = 0.0024m<sup>2</sup>).

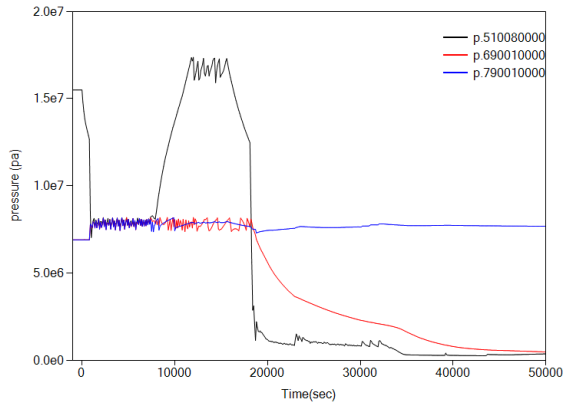


Fig. 2. Pressure of C-SGTR (c510-PZR, c690-Ruptured SG, c790-Intact SG)

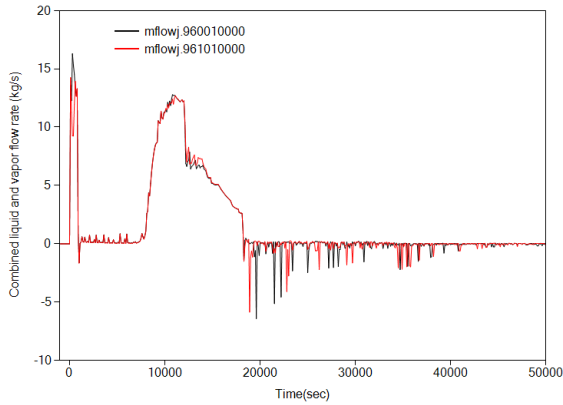


Fig. 4. Mass flow rate at U-tube break for S-SGTR (Double ended guillotine break)

## 3. Conclusions

Present study simulates the two type of SGTR using CINEMA code and reports the major reactor thermal-

hydraulics. Based on these preliminary analysis, bypass source term is going to be evaluated.

Table I. Comparison of major event

|                       | C-SGTR [s] | S-SGTR [s] |
|-----------------------|------------|------------|
| <b>RX-TRIP</b>        | 0.0        | 800        |
| <b>MSIV-TRIP</b>      | 0.0        | 900        |
| <b>MFW-TRIP</b>       | 0.0        | 900        |
| <b>RCP-TRIP</b>       | 0.0        | 12000      |
| <b>MSSV open</b>      | 20         | 900        |
| <b>SGTR start</b>     | 13580      | 0.0        |
| <b>SG dryout</b>      | 1400/1400  | 5700/5700  |
| <b>SAMG entry</b>     | 9660       | 16300      |
| <b>SDS open</b>       | 16860      | 18100      |
| <b>Vessel failure</b> | 43820      | 43500      |

## REFERENCES

- [1] J.Song et al., An analysis on the steam generator tube rupture events with core damage, Annals of Nuclear Energy, 150, 2021, 107877
- [2] J.Song et al., An analysis on the consequences of a severe accident initiated steam generator tube rupture, Nuclear Engineering and Design, 348, 2019, 14-23

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