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# APR1400 Steam Generator Tube Rupture Accident Analysis using KNAP

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## 1. Introduction

Korea Electric Power Research Institute(KEPRI) has been developed safety analysis methodology for non-loss of coolant accident(Non-LOCA) analysis of Optimized Power Rector 1000(OPR1000, previously KSNP).

The methodology has been developed using RETRAN code of Electric Power Research Institute(EPRI) as a system analysis code and named Korea Non-LOCA Analysis Package(KNAP).

Steam Generator Tube Rupture(SGTR) accident is one of the decrease in reactor coolant system inventory events and the results are typically described in the safety analysis report(SAR) chapter 15.X.

KEPRI has been analyzed OPR1000 SGTR accident analysis as a part of the unified safety analysis computer code development project and applied the methodology to Advanced Power Reactor 1400(APR1400) to confirm the feasibility of that.

APR1400 has been designed to generate about 1,400MWe with advanced features for greatly enhanced safety and economics goals. The SGTR analysis in APR1400 Standard Safety Analysis Report(SSAR) is simulated by CESEC-III code of Combustion Engineering(CE).

In this study, to estimate the feasibility of the KNAP methodology and code system, SGTR accident is analyzed using RETRAN code and it is compared those from APR1400 SSAR.

## 2. Methods and analysis results

### *2.1 The RETRAN nodalization and tube rupture model*

The standard nodalization of APR1400 is as shown in Figure 1. The basedeck includes one reactor vessel, one pressurizer and 2 separate reactor coolant system(RCS) loops with 1 hotleg and 2 coldlegs per loop. Each loop contains 2 reactor coolant pumps and 1 steam generator.

The U-tube section of the steam generator primary side is divided into 12 volumes. The secondary side of the steam generator is modeled using 14 volumes. Four main steam lines with total of 20 MSSVs are modeled.

For a tube rupture simulation, a fully single u-tube per steam generator at RETRAN nodalization is added respectively. An U-tube rupture model consists of three volumes, six junctions, three valves and trip signals for SGTR accident.

Generally, a break location in the SGTR accident analysis is assumed at the hotleg side of the loop that does not contain the charging flow, since the hotter coolant would be released more easily at the lowest downcomer temperature.

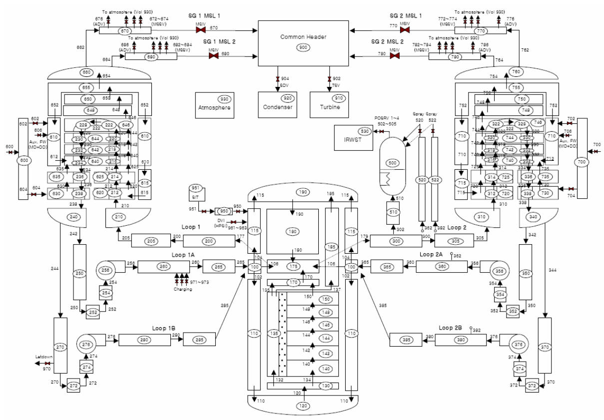

Figure 1. RETRAN Nodalization for APR1400

### *2.2 Initial conditions and assumptions*

Initial conditions for the SGTR analysis are chosen to maximize the primary coolant releases to the atmospheres during the SGTR transient.

Thus, initial conditions and assumptions are as follows : maximum core power, maximum core inlet temperature, maximum RCS pressure, and maximum pressurizer liquid volume, maximum steam generator liquid volume, minimum core flowrate.

The SGTR event occurs in the righthand SG side of the nodalization (Figure 1) and transient calculation time for analysis is assumed as 30 minutes.

When the SGTR accident starts, one valve in normally flowing junction closes and other two valves open simultaneously. Critical flow model for ruptured u-tube is selected the Extended Henry and Moody model. And the u-tube rupture is conservatively assumed to be a full double ended break.

Because a principal purpose in this study is comparison for thermal-hydraulic trends and analysis results, loss-of-offsite power(LOOP) is not applied to the initial conditions.

### *2.3 Results comparison for RETRAN and CESEC-III codes*

Firstly, the fully double-ended rupture at a single tube happens and the reactor trip signal occurs by high level trip signal of the affected steam generator.

Following the generation of Main Steam Isolation Signal(MSIS) on steam generator high level, the Main Steam Isolation Valves(MSIVs) and the Main Feedwater Isolation Valves(MFIVs) close.

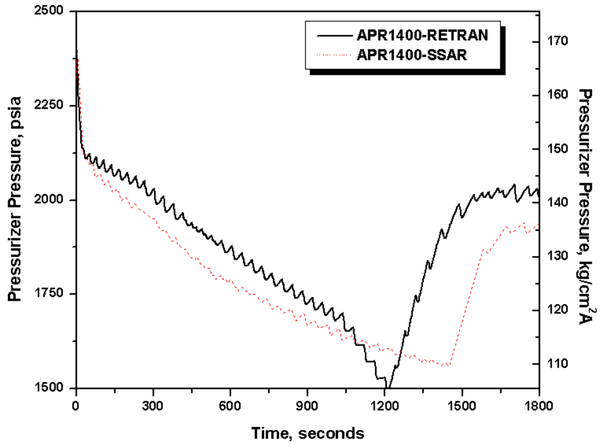


Figure 2. Pressurizer pressure vs time

After reactor trip, the RCS pressure decreases rapidly and a Safety Injection Actuation Signal(SIAS) is generated on low pressurizer level.

At the Figure 2, results of pressurizer pressure for APR1400-SSAR decrease much more rapidly than pressure for APR1400-RETRAN, this is owing to characteristics of pressurizer model within each code.

Safety injection actuation signal occurs about 1,370 seconds at the APR1400-SSAR and about 1,200 seconds at the APR1400-RETRAN respectively.

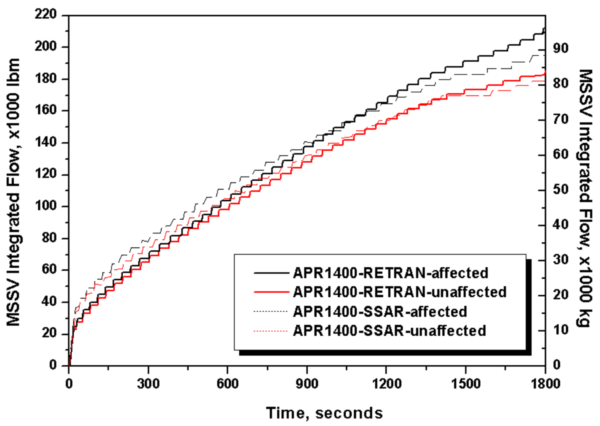


Figure 3. Integrated MSSV flow vs time

The maximum pressure of steam generator is value 87.60 $kg/cm^2A$ at the APR1400-SSAR and 87.11 $kg/cm^2A$ at the APR1400-RETRAN.

Subsequent to this peak in the pressure, the secondary system pressure decreases, resulting in the closure of the MSSVs temporarily. However, in the absence of feedwater flow due to an MSIS on the steam generator high level at the initiation of the accident, the MSSVs cycle repeatedly open and closed to remove decay heat during transient analysis.

Figure 3 gives the integrated MSSV flow versus time for SGTR event. At 1800 seconds, no more than 84,096 kg (81,734kg at the APR1400-SSAR) of steam release from the unaffected steam generator and 96,116kg (88,171kg at the APR1400-SSAR) from the affected steam generator at the RETRAN code are discharged through MSSVs respectively. It shows that steam release of RETRAN code is larger than that of APR1400-SSAR by about 9 percent.

During the SGTR event, approximately 36,759kg of primary system fluid for APR1400-RETRAN and 32,494kg for APR1400-SSAR are leaked to the affected steam generator respectively.

## 3. Conclusion

The KNAP methodology is applied to APR1400 SGTR analysis and the results are compared with those mentioned in APR1400. The thermal hydraulic behavior of two codes shows a similar trend except a some slight difference during the transients.

The maximum RCS and secondary pressures do not exceed 110% of design pressure following a SGTR accident, thus, assuring the integrity of the RCS and secondary system.

Eventually APR1400 SGTR analysis using KNAP methodology shows the acceptable results and explains the feasibility of the RETRAN code system.

In the future study, a more detailed sensitivity analysis for principal parameters and an estimation of radiological effect will be carried.


## Acknowledgements

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