Effects of Split Core Model for Steam Generator Tube Rupture Evaluation

Koh Jae Hwa, Lee Dong Hyuk, Kim Yo Han, Sung Chang Kyung
Korea Electric Power Research Institute, Safety Analysis Group, Daejeon, KOREA, euron@kepri.re.kr

1. Introduction

As a part of the unified safety analysis computer code development project funded by the Ministry of Commerce, Industry & Energy(MOCIE), Korea Electric Power Research Institute(KEPRI) has been developing the methodology of new safety analysis for Korea Standard Nuclear Power Plants(KSNP).

The pressure and enthalpy of primary system is affected by the leakage owing to tube rupture and a thermal-hydraulic behavior in the reactor vessel is important in the light of reactor heat removal and recirculation of loop flow.

To review effects for one channel or split core model in the SGTR accident, we considered two types of core model and analyzed using RETRAN-3D.

2. Reator core model and results

Generally, when modeling reactor core model, we mainly used one channel core model except more detailed analysis or steam line break (SLB) event. When analyzing SLB event, split core model was selected to simulate an accurate behavior of reactor core system because large amounts of steam release through break line and interaction between the intact loop and the faulted loop is not known exactly.

And it is not clearly defined which reactor core model is reasonable, when analyzing SGTR accident. So we modeled two types of reactor core model in RETRAN-3D and examined principal several variables of primary and secondary system.

2.1 Basic Model and Initial condition for SGTR accident

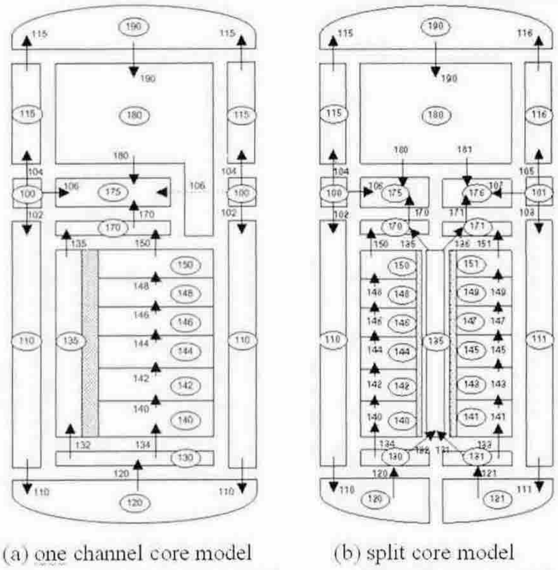
The applied nodalization and values are based on the Ulchin 3&4 units, one of the Korea Standard Nuclear Plant, and many variables are reviewed and determined for this study.

In SGTR analysis, important variables included initial core inlet temperature, initial power level, initial RCS pressure, initial pressurizer pressure and liquid level and initial steam generator liquid level.

The initial reactor operating conditions was determined to produce the most adverse consequences following a SGTR. And the analysis presented generally assumes that operator action is delayed until 30 minutes after reactor trip.

2.2 Reactor Core Model

The reactor core model is presented in Figure 1. In split model, it is necessary to consider mixing factor for energy balance with each loop. Actually, RCS flow at the faulted loop decrease due to tube leakage and steam release through the main steam safety valve. To compensate energy difference between the intact RCS loop and the faulted RCS loop, it was used appropriate mixing factor at a lower inactive core and upper inactive core part.



(a) one channel core model (b) split core model

Figure 1. Two types of reator core model for the SGTR analysis with RETRAN-3D

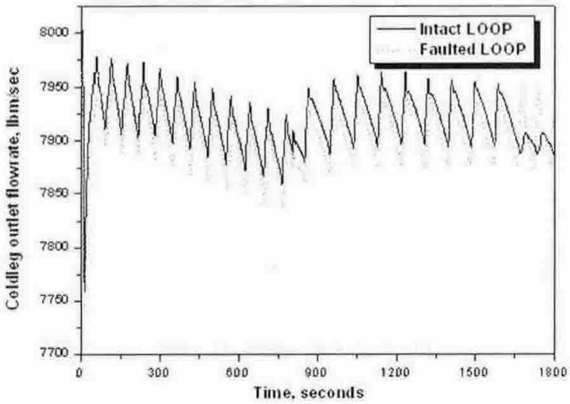
2.3 Results of Primary System

We examined the principal variables of primary system such as pressurizer pressure, pressurizer level, reactor coolant system temperature, and reactor coolant system flowrate, etc.

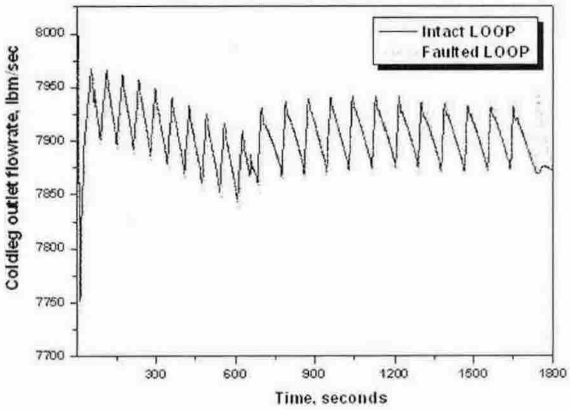
Although the results of above variables are slightly different, those are showed that the overall tendancy of primary system is same approximately.

During 1800 seconds after the steam generator tube rupture event, the flowrate into reactor vessel in the both loop are presented Figure 2.

Difference of flowrate in each model shows that a flowrate of the intact loop is more higher than that of the faulted loop due to tube leakages.



(a) one channel core model



(b) split core model

Figure 2. Flowrate into reactor vessel in the both RCS loop

In split core model, a lower flowrate of the intact loop from 1600 sec to 1800sec is represented that the MSSV remains open.

2.4 Results of Secondary System

The total amount of primary-to-secondary leakage through the tube rupture is 79,570 lbm at one channel core model and 78,350 lbm at split core model, respectively.

During the event, steam flow through the faulted steam generator main steam safety valves is 141,300 lbm at one channel core model and 138,500 lbm at split core model, while steam flow through the intact steam generator main steam safety valves is 114,900 lbm at one channel core model and 118,800 lbm at split core model.

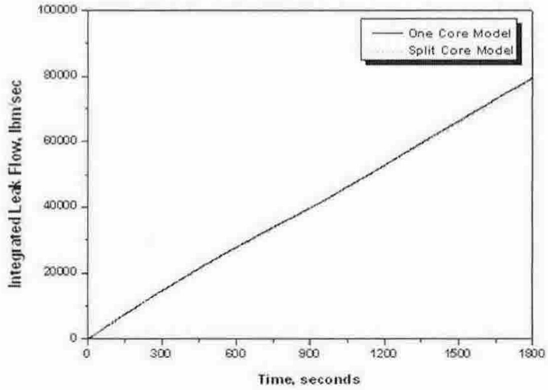


Figure 3, Integrated tube leak flow (1800sec)

Differences due to core channel model for SGTR analysis are small and can be neglected.

3. Conclusion

Through the SGTR analysis with split core model, both reactor core models show that there is a small difference of mass flowrate between intact loop and faulted loop because of tube leakage. Flowrate of the faulted loop is lower than that of the intact loop in the both core model. But one channel core model represent a more conspicuous aspect than split core model.

When considering design of reactor core model for the general SGTR evaluation, the use of either one channel core model or split core model makes no difference to analysis.

REFERENCES

[1] M.P.Paulsen et al, "RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI, July 2001.

[2] RETRAN Basedeck, "Generation of the RETRAN-3D Basedeck for Ulchin Units 3&4," FNC-CN-02-D013 Rev.03, FNC Technology Co., 2003

[3] FSAR, "Final Safety Analysis Reports, Ulchin Units 3&4," Vol.15, KEPCO