Transactions of the Korean Nuclear Society Autumn Meeting
Busan, Korea, October 27-28, 2005

Parameter Study for the Feedwater Pipe Break Analysis of an Integral Type Reactor

Young-Jong Chung[1], Soo-Hyung Kim[1], Hee-Cheol Kim[1]& Sung-Quun Zee[2]
[1]*Fluid System Engineering Division, Korea Atomic Energy Research Institute*
[2]*Reactor Engineering Division, Korea Atomic Energy Research Institute*
chung@kaeri.re.kr; ksh@kaeri.re.kr; hckim@kaeri.re.kt; zee@kaeri.re.kr

1. Introduction

Various sizes and types of advanced small and medium sized nuclear reactors are currently under development worldwide, and some of them are ready for construction [1]. The SMART, which is an integral pressurized water reactor is one of those advanced types of small sized nuclear reactors. The basic design of SMART was completed at the Korea Atomic Energy Research Institute (KAERI). In order to test and verify the SMART design, a new phase is currently underway at the KAERI [2]. The results of these tests and verifications will be fed back into the SMART basic design for a further improvement of the safety and reliability. A feedwater pipe break accident (FLB) is one of the most important accidents for the safety of the integral type reactor. Decrease in the feedwater supply to the steam generators causes a decrease in the heat extraction from the reactor coolant system, resulting in an increase of the primary coolant temperature and pressure and the nuclear power decreases due to a reactivity feedback. When assuming an actuation of the engineered safeguard systems, the core power inherently seeks a bounded level because a negative moderator density reactivity and a negative Doppler temperature reactivity are characteristics of the core design.

The calculations are performed using a system analysis code, TASS/SMR [3], developed by the KAERI. All the models including a non-condensable gas model, heat transfer model at a helical coil and a straight tube, and a drift flux model for the inter-phase change, are contained in the TASS/SMR code. The purpose of this study is to develop the most conservative case for a feedwater pipe break accident, and confirm its safety and reliability.

2. Description for the integral type reactor

The design concept of the integral type reactor is the adoption of an integral arrangement. All the primary components, which consist of a core, steam generator (SG) cassettes, main coolant pumps, and a pressurizer are integrated into a single pressurized vessel without any pipe connections between these primary components. The core is located in the lower part of the reactor vessel. While the overall arrangement of the reactor coolant system (RCS) is simplified by the elimination of the primary piping systems, the layout in the reactor vessel internal becomes more complex. The reactor coolant is forced to flow upward through the core to then flow down into the shell side of the SG cassette from the top of the SG and then back to the core.

The secondary system of the plant has four identical sections. Each of these sections can be isolated from the turbine and main feedwater control unit by the steam and feedwater isolation valves. A high and reliable level of safety is achieved through both the inherent safety core design features and the advanced engineered safety systems. Some of the safety systems are designed to function passively on demand. The core decay heat can be removed through the passive residual heat removal system (PRHRS) by a natural circulation in emergency conditions. The plant has four independent PRHRS trains with a 50% capacity for each train, and an operation of two trains is sufficient enough to remove the decay heat generated in the core. The system is capable of a decay heat removal during at least 36 hours without any action by operators for the design basis accidents.

3. Methods and results

An analysis of the feedwater pipe break accident has been performed by the TASS/SMR code [3]. The basic code structure adopts a one-dimensional geometry. The node encloses the control volumes, which represent the fluid mass and energy. The flow-path connecting the nodes represents the fluid momentum and it has no volume. The conservation variables are the mixture mass with liquid and steam, the liquid mass, non-condensable gas mass, mixture energy, steam energy, and the mixture momentum. A detailed critical heat flux ratio (CHFR) analysis is performed using a SSF-1 correlation [4], which is a one dimensional correlation for the core averaged thermal hydraulic condition.

For the conservative calculation, the following assumptions are considered in the feedwater pipe break analysis to conservatively evaluate the pressurization of the primary system. The initial conditions with different combinations are considered to identify the worst consequence. The moderator temperature and Doppler temperature reactivities, which maximize the pressure of the primary system, are selected from between the least negative value and the most negative value. A high power, high pressurizer pressure, thermal design flow, high coolant temperature, and an ANS-73 decay heat curve with the most negative Doppler temperature and the least negative coolant density reactivities are conservative. When an offsite power is unavailable, MCP pumps begin to coast down 3 seconds after a reactor trip signal occurs.

Limiting condition for an operation: The initial condition is one of the important parameters affecting the peak system pressure. It is determined that the most adverse pressure occurs when the operating condition is a high power, high coolant temperature, high pressurizer pressure and a thermal design flow with a guillotine break at the feedwater section pipe.

Reactivity parameters: The reactivity feedback is determined by the components of a moderator density, and a Doppler temperature reactivity feedback. The least negative moderator density reactivity results in a maximum pressure with the most negative Doppler temperature reactivity. Variation in the Doppler temperature reactivity has little impact on the transient since there is a small reactivity feedback under the power operation condition.

Power level: Spectra of the initial power levels are analyzed for the power levels of 100%, 75%, 50%, 36%, and 20%. Initial power level has a strong effect on the coolant temperature since it determines the time at which a core decay heat is balanced by a capacity of the passive residual heat removal system. The power level has an influence on the peak pressure. Generally, the peak pressure becomes higher when the power level is higher. This study identified a full power as the limiting power level for the spectra of the breaks occurring at the different power levels.

Steam generator heat transfer model: For the reference case, it is assumed that the heat transfer in the broken steam generator does not occur at the beginning of the transient. In this study, the heat transfer occurs at the broken steam generator. For this case, the POSRV opening time is delayed by around 1 second by a heat transfer which results from the reverse flow in the broken steam generator and the peak pressure is nearly the same as the reference calculation

4. Conclusions

A study for a conservative calculation of a FLB accident with conservative initial and boundary conditions is performed using the TASS/SMR code. A loss of offsite power is also assumed and the steam flow in the intact steam generator is assumed to be the same value as that of the broken steam generator. The largest double ended steam pipe break accident with the most reactive control rod drive mechanism in the fully withdrawn position is a limiting case under the least negative moderator density and the most negative Doppler temperature reactivity conditions and a sufficient conservatism is assured.

REFERENCES

[1] J. Kupitz, “Integration of nuclear energy and desalination systems. Proceeding of Symposium on Desalination of Seawater with Nuclear Energy (IAEA-SM-347),” Daejeon, Korea, May 26-30, 1997.

[2] S.-H. Kim, K. K. Kim, J. W. Yeo, M. H. Chung, and S.Q. Zee, “Design verification program of SMART,” Proc. Of GENES4/ANP2003, Kyoto, Japan, Sept. 15-19, 2003.

[3] H. Y. Yoon, et al., “Thermal hydraulic model description of TASS/SMR,” KAERI/TR-1835/2001, 2001.

[4] D. H. Hwang, G. W. Seo, J. C. Lee, K. K. Kim, “Development of CHF correlation systems for SMART-P fuel assembly,” KAERI/TR-2943/2005, 2005.