Sensitivity Study for a Conservative Calculation of the SLB Accident in the SMART-P Plant

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1. Introduction

A steamline break (SLB) accident is one of the most important accidents in the SMART-P for a fuel integrity point of view. Increase in the steam energy release from the steam generators causes an increase in the heat extraction rate from the reactor coolant system, resulting in a reduction of the primary coolant temperature and pressure. Assuming an actuation of the control rod or engineered safeguard systems, a core power inherently seeks a bounded level because a negative moderator density reactivity and a Doppler temperature reactivity are characteristics of the core design for the SMART-P. The calculations are performed using a system analysis code, TASS/SMR [1], developed by KAERI. All the models including a non-condensable gas model, heat transfer at a helical coil, and a drift flux model for the inter-phase change, are contained in the TASS/SMR code.

The purpose of this study is to identify the limiting cases for the SLB accident of the SMART-P, and confirming its safety and reliability.

2. Description for the SMART-P

The design concept of the SMART-P is the adoption of an integral arrangement. All the primary components, which consist of a core, steam generators, main coolant pumps, and a pressurizer are integrated into a single pressurized vessel without any pipe connections between those 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 becomes more complex. The reactor coolant is forced to flow upward through the core and then flow down into the shell side of the steam generator (SG) from the top of the SG and then back to the core.

The core decay heat can be removed through the passive residual heat removal system (PRHRS) by a natural circulation in emergency situations. The SMART-P 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. The system is capable of a decay heat removal for a minimum of 36 hours without any action by operators for the design basis accidents.

3. Methods and Results

A thermal hydraulic analysis of the SLB accident has been performed by the TASS/SMR code. The basic code structure adopts a one-dimensional geometry, and a node and flow-path network models the system responses. The node encloses 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 a mixture mass with a liquid and steam, a liquid mass, a non-condensable gas mass, a mixture energy, a steam energy, and a mixture momentum.

A detailed critical heat flux ratio (CHFR) analysis is performed using a SSF-1 (Self-sustained square finned-1) correlation [2], which is a one dimensional correlation for the core averaged thermal hydraulic condition and is calculated based on the local coolant conditions calculated at every time step.

3.1 Initial and Boundary Conditions

Conservative initial and boundary conditions are employed to analysis the SLB accident. The primary system maintains a single-phase liquid condition with a pressure of 15.5 MPa and a 103% of the nominal core power. The primary system is filled with liquid and the liquid temperature at the SG inlet and outlet is 588.08 K and 550.54 K, respectively. The secondary side of the steam generator maintains the pressure and steam temperature of 3.45 MPa and 558.61 K, respectively. The feedwater flow with 323.15 K is controlled to maintain a constant power of 103%.

The moderator density reactivity and the Doppler temperature reactivity values are selected at the most negative and least negative value, respectively. These reactivities are used in the system analysis code, TASS/SMR, to generate the system's response. The code utilizes point kinetic models to describe the core nuclear power transient initiated due to the cooldown following the SLB accident. Variations in the reactivity due to the moderator density, Doppler feedback, and rod motion are simulated in the code. An ANS-73 decay heat curve is used and the MCP pump coastdown is considered as a 3 seconds delay after a reactor trip signal occurs

3.2 Analysis Result

To determine the most conservative set of initial operating conditions and transient parameters for the SLB accident, a sensitivity analysis from a fuel integrity viewpoint is performed. These parameters include the initial pressure, initial coolant temperature, initial power, break area and location, and the power level.

Limiting condition for operation: The initial condition is one of the important parameters affecting the most adverse CHFR. It is determined that the most adverse CHFR occurs when the operating condition is a high power, high coolant temperature, high pressurizer pressure and a thermal design flow with a double ended break at the steam section pipe as shown in Fig. 1. At this point, the heat transfer in the intact steam generator is assumed to be the same value as the heat transfer of the broken steam generator. This treatment of the heat transfer increase at the intact steam generator is a basic assumption used for the sensitivity studies contained in this section.

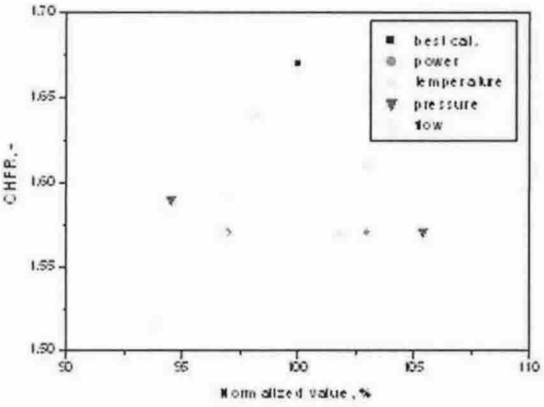

Fig. 1 the effect of the initial condition for operation

The reactor coolant flow can affect the result of the SLB accident both directly through the CHFR calculation and indirectly through the system's transient. The minimum CHFR is obtained at the thermal design flow which is the lowest possible flow. The case assuming a greater reactor coolant flow is higher than the thermal design flow case.

Power level: Spectra of the initial power levels are analyzed for the power level of 75%, 50%, 36%, 20% and 5% and a main coolant pump speed of 3600 rpm and 1300 rpm. A reactor trip setpoint by a high neutron power for the MCP high and low speed is 122% and 57.2%, respectively. The pump speed has an influence on the minimum CHFR, however the power level's effect is negligible. Generally, the minimum CHFR becomes lower when a reactor trip setpoint by a neutron power is higher. This case identifies the full power as the limiting power level for the spectra of the breaks occurring at power levels.

Break size and location: The parametric analysis includes the various combinations of the break size and steam generator heat transfer characteristics with the nominal full power plant conditions assumed. The break size is varied from 5% of the steam section pipe (ID=0.1318m) to a maximum area. The maximum area per section is limited to the sum of the steam module pipe in the steam generator cassette. The break size does not nearly have an impact on the minimum CHFR because the reactor trip setpoint by the high neutron power is the same value at a full power. A break location of the SLB is the outside of the containment because the calculation is performed as an analysis for a fuel integrity point of view.

4. Conclusion

The sensitivity study for a conservative calculation ofthe SLB accident in the SMART-P plant is performed using the TASS/SMR code. The conservative initial and boundary conditions and assumptions are as follows: the initial condition is a thermal design flow, high system pressure, high coolant temperature, anda high core power. It is assumed that a loss of the offsite power is unavailable and the heat transfer in the intact steam generator is assumed to be the same value as the heat transfer of the broken steam generator.

The largest double ended steamline break accident at the most moderator density and the least Doppler temperature coefficients conditions with the most reactive control rod drive mechanism in the fully withdrawn position is a limiting case and a sufficient conservatism is assured.

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