Transactions of the Korean Nuclear Society Autumn Meeting
Yeosu, Korea, October 25-26, 2018

Overpressure Protection Analysis for an advanced integral reactor, SMART

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1. Introduction

SMART (System-integrated Modular Advanced ReacTor) is an integral type PWR, which contains core, reactor coolant pumps (RCPs), steam generators (SGs) and pressurizer (PZR) within a single reactor vessel. SMART adopts inherent passive safety systems and is designed to be maintained at safe shutdown condition for at least 72 hours without operator's action following the initiation of any transient.

The nuclear reactor facility is required to satisfy the overpressure protection criteria in the KEPIC MNB 7000 [1] and AMSE Sec. III [2]. The overpressure protection features are required to prevent the reactor coolant system (RCS) and secondary system pressures from exceeding 110% of their design pressures.

Thus, the overpressure protection evaluation is conducted in order to examine the suitability of overpressure protection features in SMART. The protection features include the pressurizer safety valves (PSVs) for the overpressure protection of the RCS, the PRHRS safety relief valve for the overpressure protection of the secondary system, the reactor protection system (RPS) and the low temperature overpressure protection (LTOP) valve for the overpressure protection at low temperature. Among these features, the PSVs provide the most direct means of overpressure protection.

Therefore, this study focuses on the verification of the overpressure protection of SMART and the suitability of the PSV capacity during the most severe pressure transients, loss of external load event with a delayed reactor trip.

2. Analysis Methodology

The overpressure protection evaluation is conducted using the TASS/SMR-S program. The TASS/SMR-S program, developed for the safety and performance analyses of SMART, simulates the thermal-hydraulic behavior of the system by solving the conservation equations on liquid mass, mixture mass, non-condensable gas mass, mixture energy and mixture momentum.

As mentioned above, it is required to satisfy the overpressure protection criteria stated in the KEPIC MNB 7000 [1] and ASME Sec. III [2]. The pressure must not exceed 110% of the design pressure, for both RCS and secondary systems even during the most limiting transients in pressurization perspective.

Design basis event for overpressure protection analysis is the most limiting event in terms of maximum RCS pressure. In order to perform the overpressure evaluation, the design basis event is determined among the safety related design basis events which can induce an increase in RCS pressure.

In this evaluation, the loss of external load event with a delayed reactor trip was calculated to be the most limiting event in terms of maximum RCS pressure when compared to other events such as loss of normal feedwater, malfunction of the pressurizer level control system and withdrawal of the control rod assembly (CRA). Thus, the loss of external load event with a delayed reactor trip is selected as the design basis event.

Various initial conditions combining the thermal hydraulic limiting condition for operation (LCO), fuel properties and kinetics parameter are analyzed in order to determine the most conservative initial conditions. The most limiting initial conditions are determined as follows:

- High core power
- High core inlet coolant temperature
- High pressurizer pressure
- Low RCS flow rate
- Low main steam line pressure
- High pressurizer level
- Bottom skewed axial power shape
- High fuel density
- Low fuel gap thermal conductance
- Flat radial power distribution
- Least negative Doppler coefficient
- Maximum delayed neutron fraction and prompt neutron lifetime

Furthermore, following assumptions have been made for a conservative analysis:

- All control systems are not credited.
- Loss of offsite power (LOOP) is assumed to occur simultaneously with the initiating event.
- Feedwater flow is ceased simultaneously with LOOP.
- RCPs begin coastdown simultaneously with LOOP.
- Three RPS trip signals prior to the high pressurizer pressure signal are bypassed.

The selected design basis event is evaluated with the most limiting initial conditions determined above with the conservative assumptions as described in order to ensure that the acceptance criteria is satisfied. Finally, the sensitivity study on the PSV size is conducted to evaluate the PSV capacity.

3. Results and Discussion

The key focus of the overpressure protection analysis is to check whether the maximum RCS pressure is maintained within 110% of the design pressure.

In the design basis event, the loss of external load causes simultaneous LOOP. This then causes an instant feedwater loss and coastdown of the RCPs. The coastdown of the RCS flowrate can be observed in Figure 1. Passive residual heat removal actuation signal (PRHRAS) is generated by the low feedwater flow setpoint. The cooling of the RCS is solely conducted by the passive residual heat removal system (PRHRS).

In reality, the loss of external load event will be terminated by various reactor trips prior to the high pressurizer pressure trip signal. However in this analysis, the trip signals prior to the high pressurizer pressure trip signal are bypassed. The bypassed signals include low feedwater flow rate trip signal, low RCP speed trip signal and low RCS flow trip signal. The reactor is tripped by the high pressurizer pressure trip signal at 10.29 seconds as can be seen in Figure 2.

As can be seen in the figure, the PSV opens at 12.81 seconds and RCS pressure reaches its maximum at 13.04 seconds and decreases rapidly due to the PSV actuation. As can be seen in Figures 2 and 3, the maximum RCS pressure and the secondary system pressure are maintained within 110% of the design pressure. The blowdown flow through the PSV during the event is depicted in Figure 4.

Moreover, the designed PSV size can lower the pressure to its closing setpoint and safely normalize the RCS pressure. But, the capacity of the PSV has a slight impact of the pressure variance as can be seen in Figure 5.

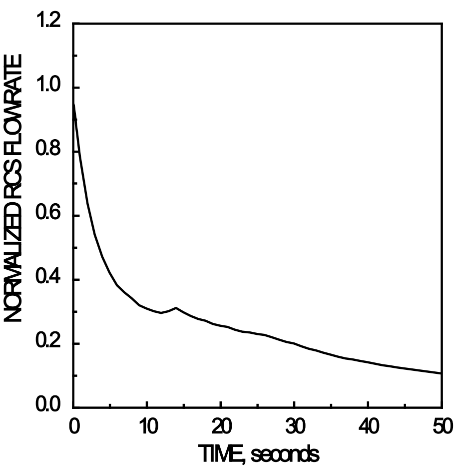

Fig. 1. RCS flowrate during loss of
external load event with a delayed reactor trip

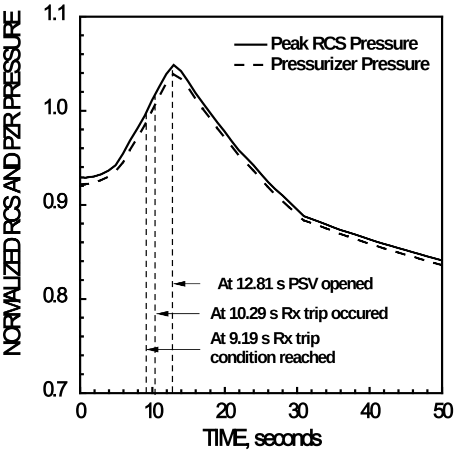

Fig. 2. Pressures of RCS and PZR during
loss of external load event with a delayed reactor trip

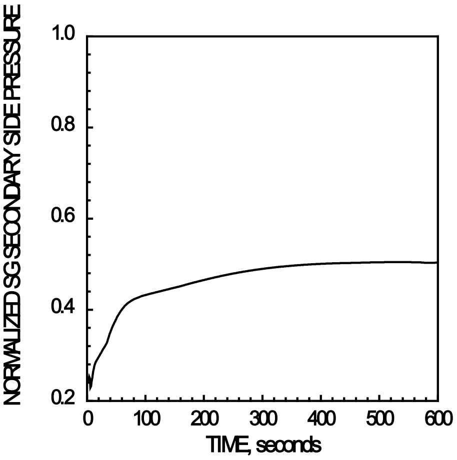

Fig. 3. SG secondary side pressure during
loss of external load event with a delayed reactor trip

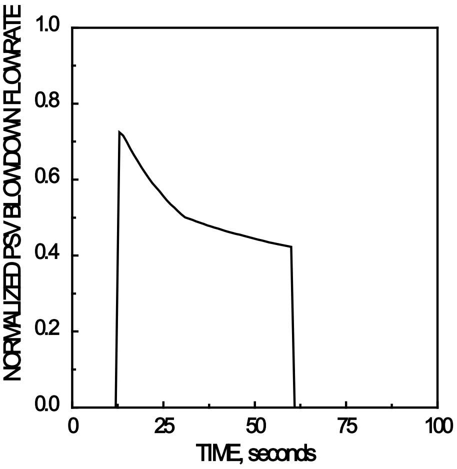

Fig. 4. PSV blowdown flowrate during
loss of external load event with a delayed reactor trip

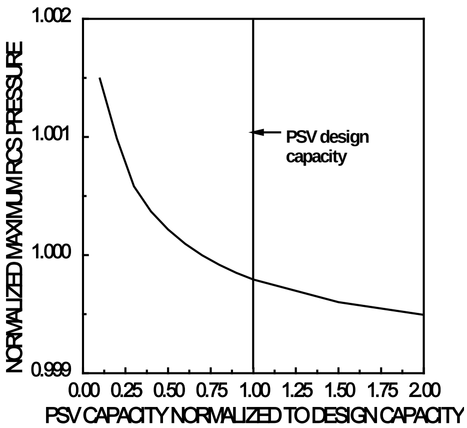

Fig. 5. Sensitivity study on PSV capacity

4. Conclusion

Overpressure can occur as a consequence of multiple scenarios. In order to examine the system integrity of the SMART design, loss of load event with a delayed reactor trip has been selected and analyzed as the design basis event for overpressure protection analysis. Conservative initial conditions and assumptions have been made for the analysis.

The results show that the RCS and secondary system satisfy the requirements in KEPIC MNB 7000 [1] and ASME Sec III [2]. The maximum pressure in the RCS and secondary system remains below 110% of the design pressure during the most severe event.

5. Acknowledgement

This work was supported by the National Research Foundation of Korea (NRF) funded by the Korea government (MSIT) (2016M2C6A1930041), in addition to funding from King Abdullah City for Atomic and Renewable Energy, Kingdom of Saudi Arabia, within the SMART PPE Project.

REFERENCES

[1] Overpressure Protection for Nuclear Reactor Facility Class I, KEPIC MNB 7000, -2000, KEPIC

[2] ASME Boiler and Pressure Vessel Code, Sections III, Rules for Construction of Nuclear Facility Components, 2017