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Preliminary Evaluation of the System Design for the Large-water-leak Accident in Prototype Generation IV Sodium Cooled Fast Reactor

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1. Introduction

Prototype Generation IV Sodium Cooled Fast Reactor (PGSFR) is a developing nuclear reactor which uses sodium as a reactor coolant and has been being designed to produce 150 MWe of electric power [1, 2]. PGSFR has many advantages such as dramatically decrease of the disposal area, the management duration, and the quantity of radioactive material when spent fuels from light water reactors are recirculated by pyro processing and burnt in PGSFR [1]. However, it has to overcome sudden pressure increase by a sodium-water reaction (SWR) accident which is occurred with tube break in a steam generator. Therefore, a protect system against the SWR should be designed.

In this paper, we investigated preliminary evaluation of the system design to drain sodium inside the steam generator for the large-water-leak accident in PGSFR.

2. Methods and Results

2.1 Sodium-Water Reaction Pressure Relief System

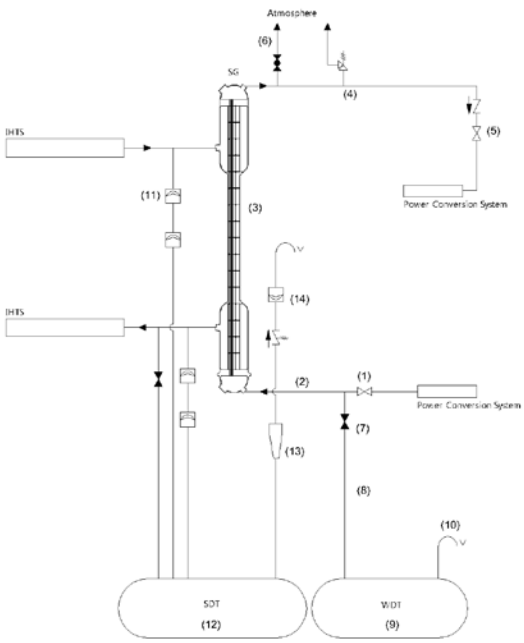

Fig. 1. Schematic diagram of SWRPRS (1) feedwater, isolation valve, (2) feedwater line, (3) steam generator, (4) main steam line, (5) main steam isolation valve, (6) atmospheric dump line, (7) water dump line isolation valve, (8) water dump line, (9) water dump tank, (10) water dump tank steam vent line, (11) main rupture disk, (12) sodium dump tank, (13) separator, (14) backpressure rupture disk [3].

Sodium-water reaction pressure relief system (SWRPRS) is a secondary subsystem connected to the intermediate heat transport system (IHTS) and designed to maintain of the integrity of the primary heat transport system (PHTS) against excessive pressure by the sodium-water reaction in the steam generator [1]. SWRPRS consists of the components such as rupture disks, a sodium dump tank, a water dump tank, a separator (Fig. 1).

2.2 Evaluation Method

Evaluation of the SWRPRS was carried out with FloMASTER™ Ver. 7 which is a commercial one dimension computational fluid dynamics software. For evaluation of the system, we used the elements such as Advanced Tanks, Storage Tank, Burst Disk, and Elastic Pipes to model our system components (steam generator, sodium dump tank, and rupture disk) and pipes connecting the components. Evaluation conditions were listed up in ref. 1. Burst pressures of the main rupture disks and the backpressure rupture disk were decided as 10.0 bar and 3.0 bar, respectively [1].

2.3 Calculation Results.

Sodium inside the steam generator was entirely drained within 42 s (fig. 2). Drain velocity of sodium through the cold reg was about 8.2 m/s in the early stage when hydrogen produced by the SWR was not vented through the hot leg pipe. After then, it was decrased from 2.5 m/s to 0.5 m/s.

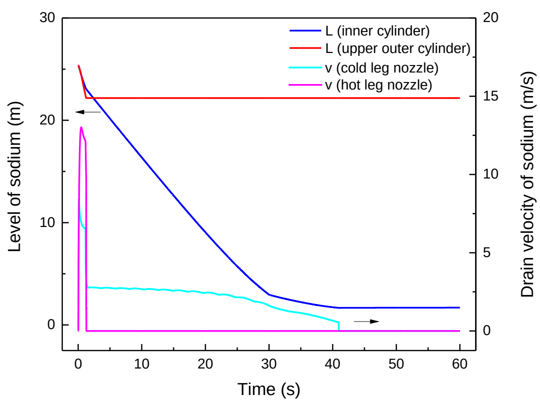

Fig. 2. Level and drain velocity of sodium inside the steam generator.

Pressure inside the steam generator increased up to 24.4 bar in 0.05 s (fig. 3). It gradually decreased to 0.14 bar for 1.4 s. When hydrogen inside the steam generator started to vent through the hot lag pipe at 1.1 s, pressure inside the steam generator dramatically reached to pressure of the sodium dump tank.

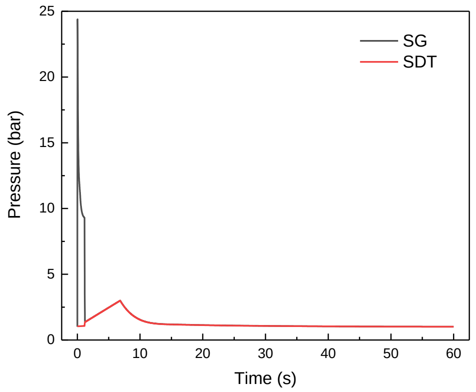

Fig. 3. Pressure inside the steam generator and the sodium dump tank.

3. Conclusions

We investigated evaluation of the system design to drain sodium inside the steam generator for the large-water-leak accident in PGSFR. Sodium was drained within 1 min and the pressure of the steam generator was below 10 bar after pressure spike (~0.1 s, ~25 bar). It was found that our current system design showed appropriate performances to drain sodium inside the steam generator when a SWR occurred by the large-water-leak accident.

ACKNOWLEDGMENTS

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