Experimental study of impingement wastage caused by sodium water reaction in the Printed Circuit Steam Generator

Siwon Seo [ab], Haeun Noh [a], Sangji Kim[c], Jaeyoung Lee [a*]
[a]*School of Mechanical and Control Engineering, Handong Global Univ., Pohang, 37554, Korea*
[b]*Atomic Creative Technology Co., Ltd., 1434 Yuseong-daero Yuseong-gu, Daejeon, 34101, Korea*
[c] *Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseoung-gu, Daejeon, 34057, Korea*

[*]*Corresponding author: jylee378@gmail.com*

1. Introduction

The sodium-cooled fast reactor (SFR) is one of the most promising Generation-IV nuclear reactors. The SFR is liquid metal reactor using sodium as a coolant. Typically, shell and tube type steam generator have been applying to the SFR. Two different fluids are flowing in this component. These are liquid sodium and water (Figure 1). It means that the SFR has inherent risk in steam generator. That is sodium-water reaction (SWR). The SWR is inevitable problem in the SFR as long as water is used as secondary fluid in the SFR steam generator (SG).

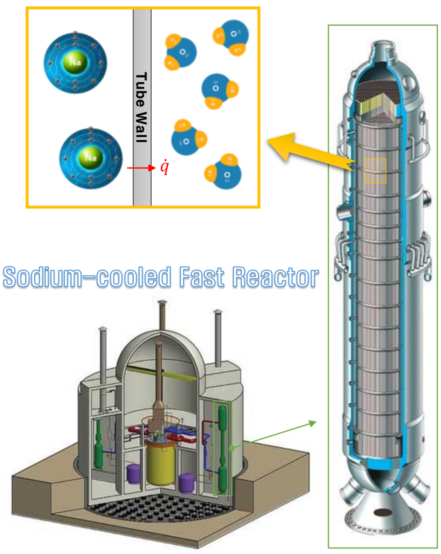

Fig. 1. Typical Steam Generator used in the SFR

If heat transfer tube between liquid sodium and water is ruptured by any reason, sodium and water will contact with each other. Then exothermic chemical reaction will occur according to mainly the following chemical formula (Baldev Raj *et al*, 2015).

$$Na + H_2O \rightarrow NaOH + 1/2H_2 - 147.37 \text{ kJ/mol}$$

If SWR is happened, SG integrity could be degraded by reaction product, NaOH, due to its corrosive property. This phenomenon is called wastage. If corrosion and erosion occur at failure part on heat transfer tube of SG, it is called self-wastage. And if they occur at opposite heat transfer tube, it is called impingement wastage (Figure 2). The most limiting case of the SWR accident occurred in shell and tube SGs is multiple tube failure by impingement wastage.

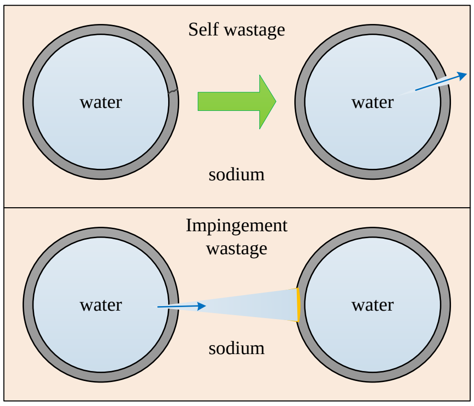

Fig. 2. Schematic Drawing of wastage phenomena

In nuclear safety, prevention and mitigation are basic strategy to deal with nuclear power plant accident. Prevention or mitigation way of the SWR accidents has to be established for stable and safe operation of the SFR. For this, many safety researches such as double walled SG tube[1], detection of SWR[2~4], self and impingement wastage in shell and tube SG[5~11] had been performed. Brayton cycle is also used to prevent SWR using compact heat exchanger[12]. In this study a Printed Circuit Steam Generator (PCSG) is used for mitigation the SWR accident. The PCSG is a kind of compact heat exchangers. It is fabricated that chemically etched steel plates are consolidated into a body by diffusion bonding (Figure 3). If the PCSG could be applied to the SFR, following safety advantages are expected.

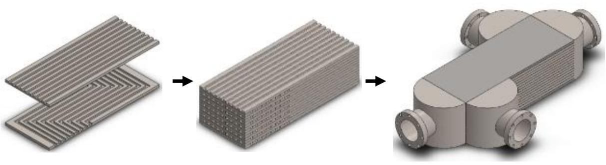
Fig.3. Fabrication procedure of the PCSG

- There is no damage propagation by impingement wastage due to very short target distance
- Effective accident management could be possible by modularization of the PCSG

- Acoustic detection of the SWR could be effectively applicable due to low background noise caused by laminarization of flow in the PCSG

Experimental studies are performing now to show these predicted advantages of the PCSG, especially no damage propagation caused by impingement wastage. In this paper, experimental works about measuring quantitative impingement wastage rate will be presented.

2. Methodology

2.1 Design and Fabrication of experimental apparatus

As shown in Figure 4, the PCSG in the SFR consists of a sodium channel and a water channel in each layer. The left side of Figure 4 is a conceptual diagram of a PCSG, and the right side is an internal structure of the PCSG. Red and blue represents the sodium channel and the water channel, respectively. And yellow represents discharged steam jet into the sodium channel.

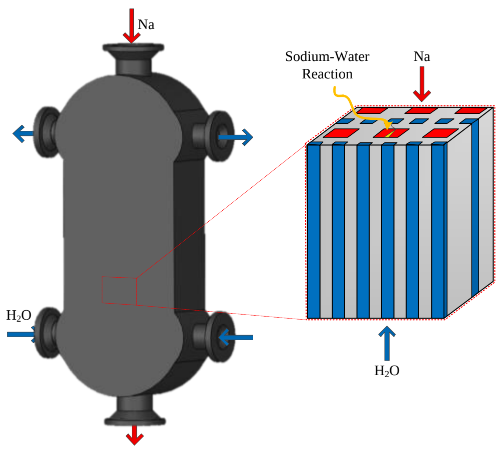

Fig. 4. Schematic of the PCSG and cross-sectional area

The pressure on the water side of the PCSG is about 180 bars whereas the pressure on the sodium side is almost atmospheric pressure. That is, as water is discharged into the sodium channel due to a considerable pressure difference, a sodium-water reaction may occur in a narrow sodium channel (about 4 mm in diameter) of the PCSG. To simulate SWR in the PCSG, a test section was designed and fabricated as a Figure 5. The vertical channel of the test section simulates the sodium tube, the horizontally connected channel simulates the water channel, and there is a very small hole between the sodium channel and the water channel, which simulates the crack between the two channels. A rupture disc was installed between the water and the sodium channel to control the isolation and contact time of sodium and water.

The experimental apparatus consists of water supply system, sodium supply system, vacuum system, and filtered venting system. The water supply system and sodium supply system supply water and sodium to the test section, respectively. A vacuum system removes oxygen and moisture in the sodium pipe to prevent sodium fire and oxidation. A filtered venting system prevents overpressure in the experimental apparatus. Figure 6 is a schematic diagram of the CATS-S facility.

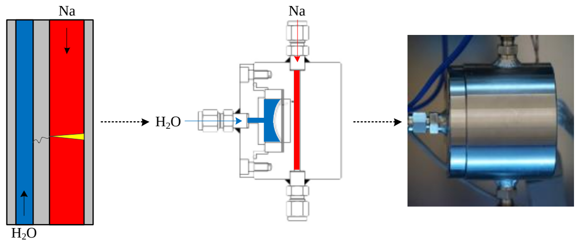

Fig. 5. Test section modeling and design

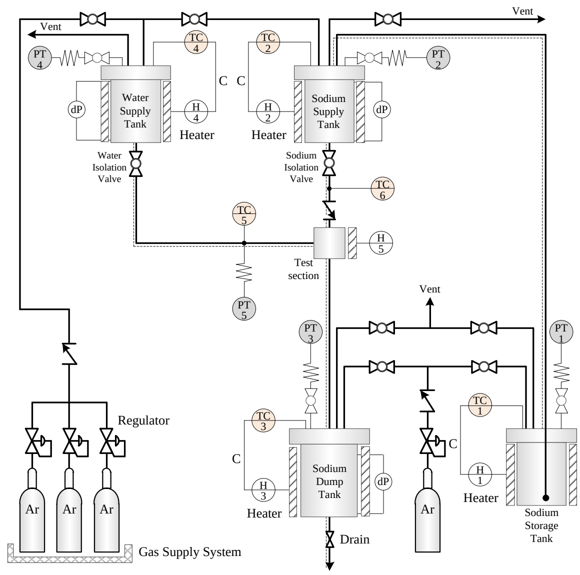

Fig. 6. P&ID of the CATS-S (Compact Accident Tolerance Steam Generator – SWR)

2.2 Experimental Conditions

It is reported that almost SG tube failures in SFR had occurred at welding area between tube sheet and U-tube. In practice, total five SWR accidents occurred in Phenix during its life time[13]. First four accidents are happened by fatigue crack at welding area. However, the PCSG is different from the conventional SG. There are no research results about the SWR in the PCSG. So there is no report about possible location of the SWR in the PCSG. Theoretically, possible reason of the boundary failure between sodium and water channel is stress by high pressure difference. Stress formula used to estimate mechanical integrity of the PCSG is as follow [14].

$$E=\Delta P\ (1/(N\cdot t)-1) \qquad (1)$$

E is stress exerted on heat exchanger. ΔP is pressure difference between two fluids. N and t are the number of channel wall per unit length and wall thickness between two fluids, respectively. Minimum wall thickness and pitch for maintaining mechanical integrity under operating condition can be derived by using above formula. Maximum allowable stress is fixed according to materials. N and t are also fixed value by design of the PCSG. It means possibility of the failure between sodium and water channel is maximized in maximum pressure difference region of the PCSG. Typically, the most high pressure difference is applied at water inlet region of SGs in the SFR. Generally, operating pressure of the water inlet and sodium outlet are approximately 180 and 1 bar, respectively. Therefore, operating conditions of the water inlet region of the PCSG will be considered as test conditions of this experiment. Operating conditions of the PCSG are presented in table 1.

Tab. 1. Operating conditions of the PCSG

Input Parameters	Water side	Sodium side
Press. (MPa)	18 ~ 16.7	0.5
Temp. (℃)	240 ~ 503	528 ~ 332

Among the above operating conditions, the pressure on the water side was too high to be simulated in the laboratory, so the experiment was conducted by lowering the pressure based on important phenomenon. When water is discharged from the water side into the sodium channel, the most important phenomenon is the critical flow phenomenon. The water pressure at which critical flow can occur was calculated and the experiment was performed at a pressure greater than that pressure, and the leak rate was changed by controlling the water side pressure. Up to now, as shown in Table 2, experiments have been performed for two cases, and the difference between the two cases is the leak rate and sodium temperature.

Tab. 2. Experimental Conditions

Test No.	Water Side Press (MPa)	Water side Temp. (℃)	Sodium Side Press (MPa)	Sodium side Temp. (℃)	Hole size (I.D., mm)
Test 1	2.5	240.0	0.1	332	0.3
Test 2	3.5	240.0	0.1	450	0.3

3. Results and Discussion

The SWR experiment in PCSG was performed, and the specimen was wire cut as shown in Figure 7. The cut specimen was measured for impingement wastage rate as shown in Fig. 8 using a Laser Displacement Sensor (LDS). The experimental results for both cases can be expressed as shown in Figure 9. As a result of the experiment, Test 1 had a larger wastage rate than Test 2 even though the leak rate was smaller. It is estimated that the wastage occurred due to formation of the toroidal shape jet in Test 2. And in Test 1, it is predicted that the pit shape jet makes lager wastage than Test 2.

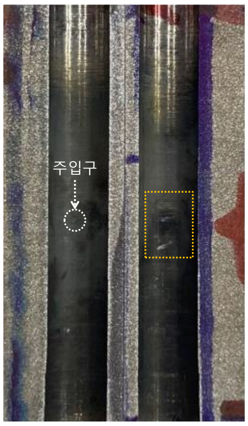

Fig. 7. Wire cut test section

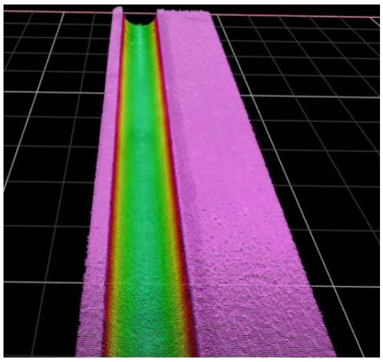

Fig. 8. LDS measurement result

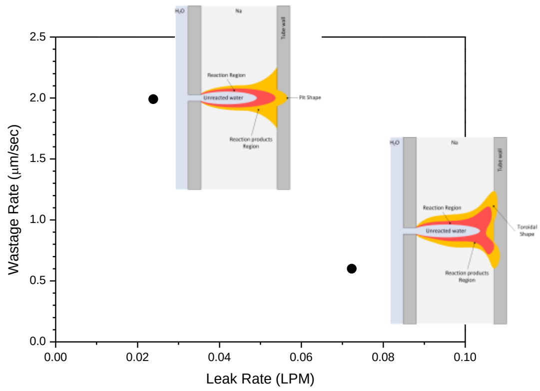

Fig. 9. Leak rate vs. wastage rate results

According to the results of wastage rate tests for stainless steel in the existing shell & tube steam generator, the maximum wastage rate occurred at a leak rate of about 2.721 LPM, and the wastage rate was 5.08 μm/sec[5]. In the PCSG environment, since the target distance is significantly smaller than that in the shell and tube SG, it is expected that the maximum wastage rate will be measured at a much smaller leak rate, and it is judged that the results are yielded as expected. It is judged that an experiment at a lower flow rate is necessary to find the leak rate that causes the maximum impingement wastage in the PCSG.

4. Conclusions

The introduction of PCSG was considered as a way to mitigate SWR accidents that may occur in SFR. The PCSG predicted that there would be no damage propagation because the wastage was smaller than that of the existing shell and tube steam generator even if SWR occurred, and an experiment was conducted to show this. Up to now, two cases of experiments have been performed, and as a result of the experiment, it is determined that the maximum wastage rate will occur at a significantly lower leak rate than that of the existing shell and tube SG. In the future, additional experiments will be carried out to find the leak rate that causes the maximum wastage rate.

ACKNOWLEDGEMENT

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIP). (No.2017M2A8A4018812)

REFERENCES

[1] Mari Marianne Uematsu et al., Comparison of JSFR design with EDF requirements for future SFR, Journal of Nuclear Science and Technology, Vol. 52, No. 3, pp.434-447, 2015.

[2] H. Nei et al., Acoustic Detection for Small leak Sodium-Water Reaction, Journal of Nuclear Science and Technology, 14(8) 558-564, 1977.

[3] Acoustic Signal Processing for the Detection of Sodium Boiling or Sodium-Water Reaction in LMFRs, IAEA-TECDOC-946, 1997.

[4] T. Kim, J. Jeong, S. Hur, Performance Test for Developing the Acoustic Leak Detection System of the LMR Steam Generator, Transaction of the KNS Autumn Meeting, 2005.

[5] H. V. Chamberlain, Project Summary – Sodium-Water Reactions Related to LMFBR Steam Generators, APDA-257, Atomic Power Development Associates, 1970.

[6] N. Kanegae et al., Wastage and self-Wastage Phenomena Resulting from Small Leak Sodium-Water Reaction, PNC TN941 76-27, Power Reactor & Nuclear Fuel Development Corporation, 1976.

[7] M. Nisimura et al., Sodium-Water Reaction Test to Confirm Thermal Influence on Heat Transfer Tubes, PNC TN9400 2003-014, Power Reactor & Nuclear Fuel Development Corporation, 2003.

[8] K. Shimoyama, Wastage-Resistant Characteristics of 12Cr Steel Tube Material, PNC TN9410 2004-009, Power Reactor & Nuclear Fuel Development Corporation, 2004.

[9] Y. Deguchi et al., Experimental and Numerical Reaction Analysis on Sodium-Water Chemical Reaction Field, Mechanical Engineering Journal Vol.2, No.1, 2015.

[10] S. Kishore et al., An Experimental Study on Impingement Wastage of Mod 9Cr 1Mo Steel due to Sodium Water Reaction, Nuclear Engineering and Design 243 (2012) 49-55

[11] S. Kishore et al., Impingement Wastage Experiments with 9Cr 1Mo Steel, Neclear Engineering and Design 297 (2016) 104-11

[12] D. Plancq et al., Progress in the ASTRID Sodium Gas Heat Exchanger development, IAEA-CN245-286. 2017.

[13] J. Guidez, G. Prele (2017). Superphenix: technical and scientific achievements.

[14] Hesselgreaves, J. E. (2001). Compact Heat Exchangers: Selection, Design and Operation (2nd), Pergamon Pres