DBE Analysis for KALIMER-600

Kwi- Seok Ha, Hae-Young Jeong, Young-Min Kwon, Won-Pyo Chang, Yong-Bum Lee, and Young-IL Kim
Korea Atomic Energy Research Institute, P.O. Box 105, Yuseong, Daegeon 305-600
E-mail:ksha@kaeri.re.kr

1. Introduction

The SFR (Sodium Fast Reactor) which is being developed at KAERI (Korea Atomic Energy Research Institute) is currently divided into three types, such as, Advanced Concept 600 MWe break-even reactor and burner reactor and 1200 MWe break-even reactor. As a part of accidents analysis of the 600 MWe break-even reactor, 5 representative DBE's (Design Bases Events) are analyzed for the safety analysis. The 5 DBE's are TOP (Transient of Over Power), LOF (Loss Of Flow), LOHS (Loss Of Heat Sink), Pipe Break, and SBO (Station Black Out).

2. Analysis of 5 DBE's

Advanced concept 600 MWe break-even reactor is a pool type sodium-cooled fast reactor with thermal power of 1538 MW and it uses metallic fuel of U-TRU-10%Zr for a core [1]. The plant has an inherent safety characteristic owing to the design to have a negative power reactivity coefficient during all operation modes and it has a passive safety characteristic due to the design of a passive decay heat removal circuit (PDRC).

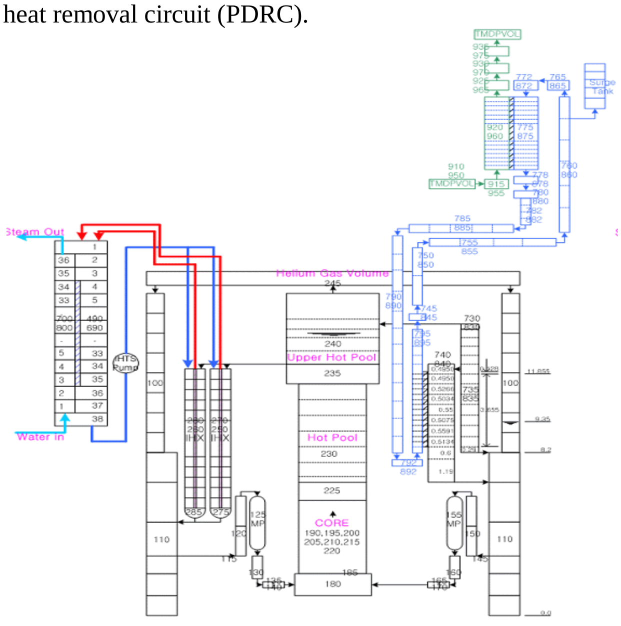

Fig. 1. MARS-LMR nodalization for SFR 600 MWe

The figure 1 shows the MARS-LMR [2] nodalization for the system. In the primary system two main pumps takes sodium from the pool and discharge it into inlet pipes. Then the flow is entered into the inlet plenum feeding fueled driver subassemblies. The sodium is heated through 7 core regions and mixed in an outlet plenum of the reactor. Then the sodium goes IHX(Intermediate Heat eXchanger) inlet through lower hot pool nodes. In the IHX, the sodium transfers its heat to the sodium of intermediate loop. The primary sodium leaving the IHX dumps directly back into the cold pool.

For the identification of these safety characteristics, 5 DBE's are analyzed using MARS-LMR codes [2]. All events are assumed to be occurred at the full power condition. The additively considered assumptions are as follows; (1) Reactor scrammed by a high power trip of 111 %, high core outlet temperature of 555 ℃, low pumping flow rate of 84 %, or low hot-pool level of 5 cm below from a normal level. (2) ANS-79 model is used for core decay power after reactor scram. (3) Pump trip is occurred at 5 seconds after reactor scram and coastdown time of pump is about 120 seconds. (4) The isolation time of SG (Steam Generator) feed water line is the same as pump trip time. (5) Two independent PDRC's are available.

The TOP accident is initiated by a possible malfunction of the reactivity controller due to control rods withdrawal and the core power is rapidly increased by a positive reactivity insertion. As shown in Fig. 2, the event occurred at 10 seconds and thereby reactor trip is occurred at 16.74 seconds by a high power trip. After the pump trip of 21.74 seconds, flow and power are simultaneously decreased and the coolant temperatures are slowly fallen at 6150 seconds by the operation of PDRC's.

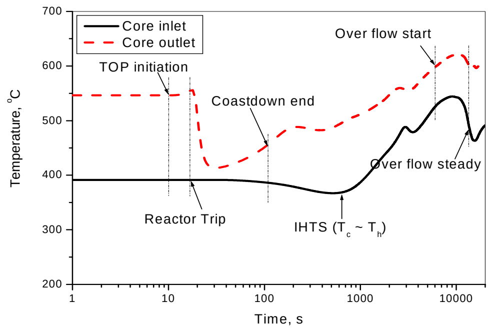

Fig. 2. Coolant temperatures behaviors at TOP

The LOF means the loss of core cooling capability due to the pumping failure of primary pumps. In this simulation, all primary pumps are tripped at 10 seconds, thereby the reactor scram is occurred at 11.84 seconds by low primary pumping flow rate. After this event development is similar to the TOP event. Fig. 3 shows the over-flow behavior from hot pool to DHX (Decay Heat eXchanger) shell of PDRC system and at 6250 seconds the coolant temperatures are decreased by the operation of PDRC's.

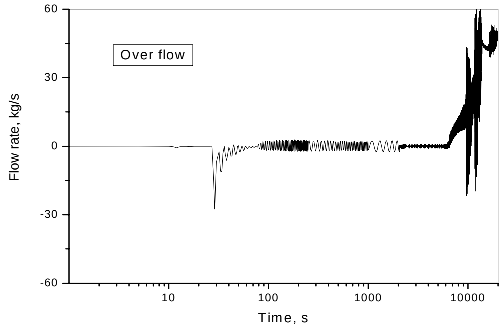

Fig. 3. Over-flow to PDRC system at LOF

The LOHS is caused by a loss of feedwater to all SG's or all pumps trip in IHTS (Intermediate Heat Transport System). In this simulation a loss of feedwater to SG's is assumed to be occurred at 10 seconds. RPS (reactor protection system) senses slowly the accident because the effect of the event transported to the RPS by way of IHTS. So the reactor scram is occurred at the 76.65 seconds. After pump trip the coolant temperatures go rapidly up and the maximum temperature of hot rod of 690 ℃ is calculated at 252 seconds. The other hand the operation of PDRC's is fast because the level of hot pool rises rapidly due to the high temperature of hot pool coolant.

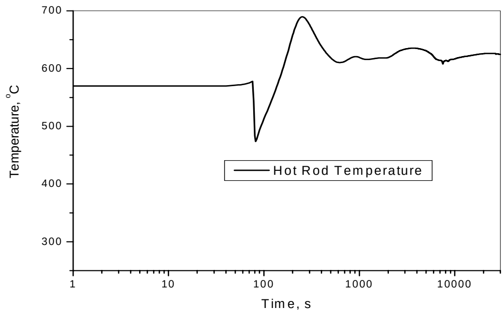

Fig. 4. Hot rod temperatures behaviors at LOHS

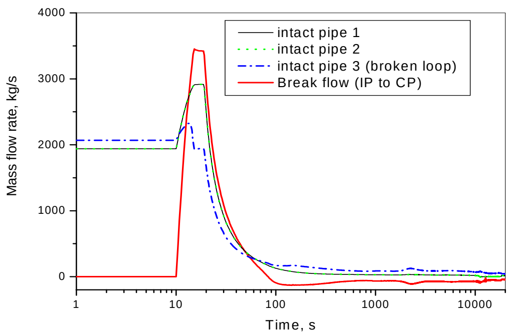

Fig. 5. Pipe flow behaviors at Pipe-break

The accident of pipe break is assumed to be occurred at between one of inlet pipes and the inlet plenum. Thus the sodium of inlet plenum through intact pipes is directly discharged into the lower cold pool. This event is initiated at 10 seconds and the rapid reduction of core flow causes the fast increase of core outlet temperature from which reactor is scrammed. Fig. 5 shows the flow behaviors in inlet pipes and inlet plenum to cold pool. After 100 seconds, the flow from the inlet plenum to cold pool is negative and this means to be formed more natural circulation flow than the flow in the normal flow path. From the viewpoint of long term cooling, this event is less severe than the other events.

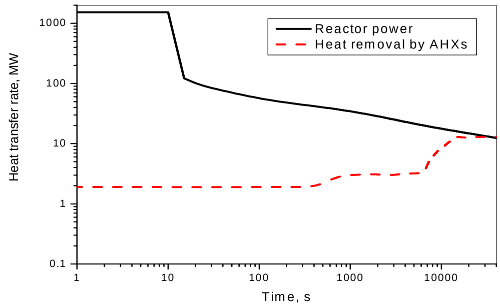

Fig. 6. Heat removal by PDRC system at SBO

The SBO is initiated at 10 seconds and so reactor scram, primary pump trip, secondary pump trip, and SG feedwater isolation is simultaneously occurs at the same time of the accident initiation. The event development after reactor scram is similar to those of TOP. As shown in Fig. 6, the heat removal by PDRC system exceeds the core decay power about 30000 seconds and the plant reaches quasi-steady state.

3. Conclusion

The 5 DBE events of SFR system transients are analyzed using MARS-LMR and all events satisfy the safety acceptance limit of 700 ℃ as shown in Table 1. The calculation results indicate that the advanced concept 600 MWe SFR has the passive safety characteristic by PDRC's.

Table 1. Summary of analysis results

Event	LOF	TOP	LOHS	Pipe Break	SBO
Reactor scram	11.84	16.74	76.65	14.06	10
FW isolation	16.84	21.74	81.65	19.06	10
PDRC operat.	6250	6150	2000	5000	5780
PCT, ℃	625	625.75	690	619	625

ACKNOWLEDGMENTS

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[2] Kwi-Seok Ha et al., "Simulation of EBR-II SHRT-17 test using MARS code", Proceeding of NUTHOS-7, seoul, 2008.