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Analysis of Design Basis Events in a Preliminary Specific Design of PGSFR

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1. Introduction

KAERI(Korea Atomic Energy Research Institute) has been developing a preliminary specific design of the PGSFR(Prototype Gen-IV Sodium-cooled Fast Reactor), which is a pool type sodium cooled fast reactor with a thermal power of 392.2 MW.

Many alterations were made on a preliminary specific design of the PGSFR compared with a conceptual design: a heat removal capability of the DHRS was decreased, the DHXs were submerged in a cold pool, a pressure drop through the core was increased, and a shape of a redan was changed to a peanut type, etc.

For identification of safety characteristics including the design changes, 5 DBE's(Design Bases Events) were analyzed using MARS-LMR code. The representative DBE's are TOP(Transient of Over Power), LOF(Loss Of Flow), LOHS(Loss Of Heat Sink), Reactor Vessel Leak and Pipe Break accidents.

2. Methods and Results

Figure 1 shows a nodalization of the PSGFR for MARS-LMR simulation. The PGSFR is composed of a Primary Heat Transport System(PHTS), an Intermediate Heat Transport System(IHTS), a Steam Generating System(SGs) and a safety-grade decay heat removal system(DHRS). The DHRS is composed of 2 units of Passive Decay-heat Removal Circuits(PDRC) and 2 units of Active Decay-heat Removal Circuits(ADRC). Each unit can remove 0.25 % of the nominal power. The ADRC can also be operated in passive mode, which corresponds to 0.125 % heat removal.

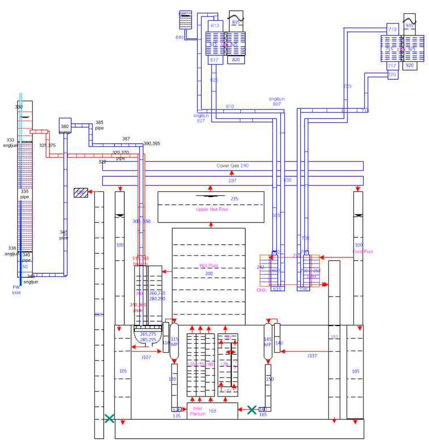

Figure 1 MARS-LMR nodalization for PGSFR

The TOP, LOF, LOHS, primary pipe break and reactor vessel leak events for the representative DBAs were analyzed using MARS-LMR code. All events were assumed to start at 102 % power. The ANS-79 model was used for a core decay power after a reactor scram. A fission product factor was conservatively considered as 1.2 times. At 5 seconds after a reactor trip, SG feed-water lines were isolated and the primary and secondary pumps were tripped. Two independent PDRC's and one ADRC were assumed to be available in accordance with a single failure criterion. AHX and FHX dampers were assumed to be open at 5 seconds after a reactor trip.

The TOP accident was assumed to be initiated due to a control rod withdrawal by a drive motor failure at 10 seconds and a positive reactivity was inserted by the amount of 30 ¢ during 15 seconds. A reactor trip occurred at 15.6 seconds by a high power/flow trip. A peak cladding temperature was calculated to be 626.7 ℃. The heat removal by three DRCs exceeded a core power after 41810 seconds and a core outlet temperature decreased continuously.

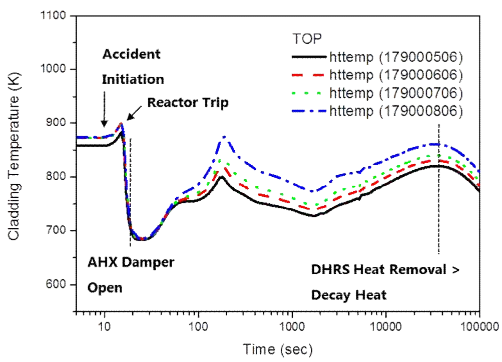

Figure 2 Cladding temperatures during TOP

A LOF means a loss of core cooling capability due to a pumping failure of the primary pumps. An imbalance between a reactor power and the primary flow is a main safety concern in the LOF event. To prevent an occurrence of the severe imbalance between power and flow, the reactor was designed to be tripped by a high power/flow trip. In this simulation, all the primary pumps were tripped at 10 seconds. A reactor scram occurred at 12.6 seconds and a reactor power and the primary flow decreased. The power decreased drastically due to the reactor trip and a cladding temperature showed the highest value. The peak cladding temperature was calculated to be 615.8 ℃.

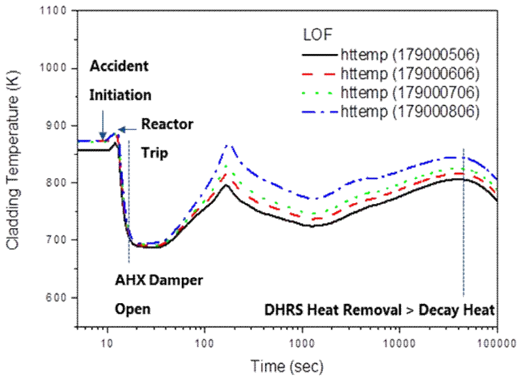

Figure 3 Cladding temperatures during LOF

The LOHS accident was assumed to be initiated from isolation of feed-water to steam generators. The IHTS pumps and PHTS pumps were also stopped on the assumption that a loss of offsite power occurs at 5 seconds after a reactor trip. Therefore, a residual heat removal was achieved by an evaporation of water in SG tubes and by the DHRS after the accident. In this simulation, a loss of feed-water to SG's was assumed to occur at 10 seconds. A reactor was tripped at 48.8 seconds by an abnormal rise of the IHX inlet temperature after the accident. After the pump trip, the primary coolant temperature rapidly went up and the maximum cladding temperature was calculated to be at 623.8 ℃.

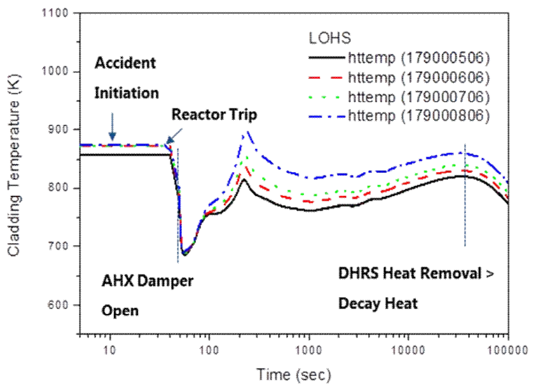

Figure 4 Cladding temperatures during LOHS

The primary coolant flows into an inlet plenum from four pipes connected with two PHTS pumps. The primary pipe break accident was assumed to be initiated from a pipe break of one of the four pipes. A flow through the broken pipe is discharged into a cold pool. The highest cladding temperature was estimated in this accident. The peak cladding temperature was calculated to be 787.1 ℃.

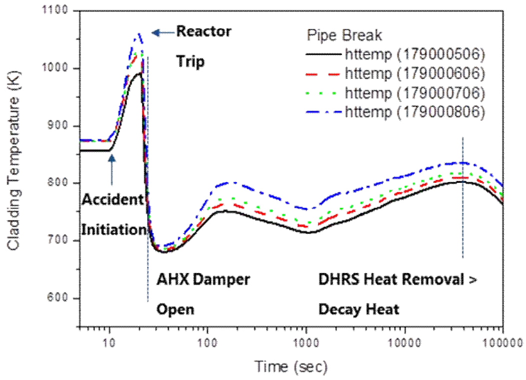

Figure 5 Cladding temperatures during Pipe Break

A reactor vessel leak accident is a typical accident of a sodium leak occurred at the PHTS boundary. It mainly affects a level of sodium in the PHTS. The leak was conservatively assumed to occur at the bottom of the reactor vessel and the leak size was assumed to be 10 cm2. The accident was assumed to occur at 10 seconds. A reactor trip occurred at 412.0 seconds. It was detected by a low sodium level after the accident. The peak cladding temperature was calculated to be 604.4 ℃.

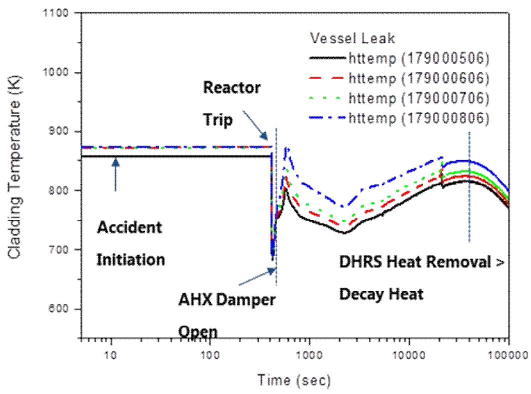

Figure 6 Cladding temperatures during Vessel Leak

3. Conclusions

The representative DBE's were analyzed using the MARS-LMR code. As a result, it was identified that the PGSFR were appropriately tripped by the RPS(Reactor Protection System) and cooled by the DHRS. But a high cladding temperature was estimated in a pipe break accident. Therefore, integrity of the structure should be evaluated in the further study.

REFERENCES

[1] K. S. Ha, Safety Evaluation for Transients of Demonstration SFR, Korea Atomic Energy Research and Institute, 2011.