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Analysis of 2.5% Reactor Inlet Header Break Accident in Wolsong-1

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1. Introduction

The Korea Institute of Nuclear Safety (KINS) is developing the integrated safety analysis system for the PHWR. The MARS-KS-CANDU is CANDU version of the MARS-KS and it has adopted several models from RELAP-CANDU such as fuel heat up, bundle CHF, CANDU header, horizontal stratification and pressure tube deformation. Many separate and integrate effect tests have been performed for PHWR, but few studies have been conducted for small break loss of coolant accident. The 2.5% reactor inlet header break accident in PHWR was assumed and analyzed in this study using the MARS-KS-CANDU. The reference plant was Wolsong-1, which is one of the CANDU type nuclear power plants operating in KOREA.

2. Description of Accident Scenario and Modeling

2.1 Accident Scenario

The Small Break Loss of Coolant Accident (SBLOCA) can cause complicate transient phenomena in the coolant system. The small break is limited to the size of the largest feeder pipe according to CNSC documents. The break of 2.5% reactor inlet header was assumed because it is a limiting case for loss of coolant inventory. When the break occurs, the coolant is discharged to containment so that the coolant inventory and total loop flow decreases. The reactor trip signal is initiated by RCS low pressure signal after 92 seconds of break. When the RCS pressure drops to 5.25MPa, LOCA signal occurs and the loop isolation motor valve is closed to isolate the broken loop. After 30 seconds of LOCA signal, the main steam safety valve opens and it leads to steam generator crash cool down. The High Pressure Emergency Core Cooling (HPECC) injection to the broken loop is initiated when the RCS pressure decreases below 4.0MPa. After 2 minutes of automatic pump trip signal, all the reactor coolant pumps are stopped and flow stagnation could occur in each core channel. The ECC flow performs a long-term cooling in the intact and broken loop.

2.2 System Modeling

The small break in the reactor inlet head was considered, where all the safety systems were assumed to be available. The primary coolant system of Wolsong-1 consists of 2 loops among which one is assumed to be broken, and another intact. Total of 380 fuel channels are modeled by 4 multi averaging channels according to power and location of each channel. One of the multi averaging channels in the broken loop is separated into 7 single channels to perform a detailed analysis for core flow. Each loop has inlet header, outlet header, steam generator and reactor coolant pump. The loop isolation valve located at the pressurizer isolates the loops in normal operation. Figure 1 shows the nodalization of Wolsong-1.

3. Results

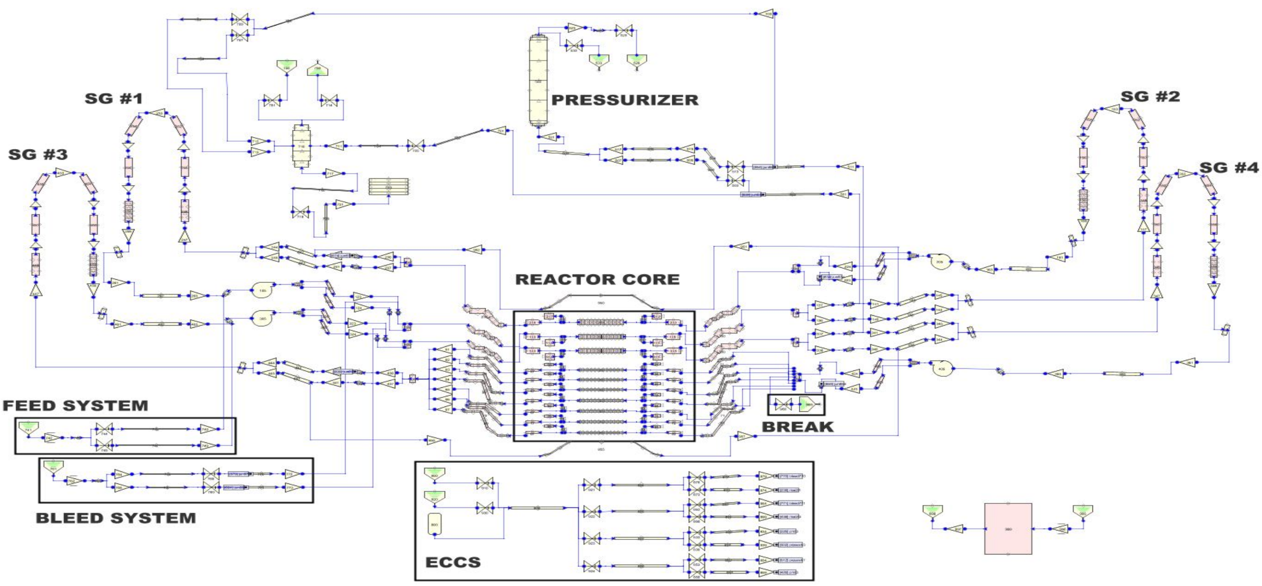

Fig. 1. Nodalization of Wolsong-1

The break of 2.5% reactor inlet header which has 5.33×10^{-3} m^2 break size causes 450kg/s of coolant and 1130kJ/kg of enthalpy discharges in the early stage of the accident. The pressure of broken loop decreases according to the discharge rate until reactor trip at 92 seconds. After the reactor trip, because the heat generation in the core is smaller than heat removal from steam generator, the pressure is sharply decreased by the shrinkage of the coolant. In the long term, the ECC injection initiated at 132 seconds mitigates the decrease of pressure. This is shown in figure 2.

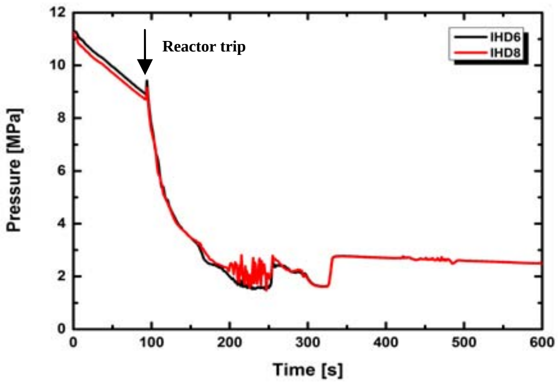

Fig. 2. Broken loop inlet header pressure

In order to compensate the loss of coolant, the coolant in the pressurizer flows into the broken loop after the break. But after the depletion of pressurizer, the core flow started to decrease rapidly by vapor generation at the pump inlet. When the ECC injection is initiated, the vapor is eliminated and the loop is refilled so that the core flow recovers its normal condition at 250 seconds. At 292 seconds, all the RCPs are stopped, so the core flow decreases again. The MPECC and LPECC supply the minimum flow to the core. Figure 3 shows this feature.

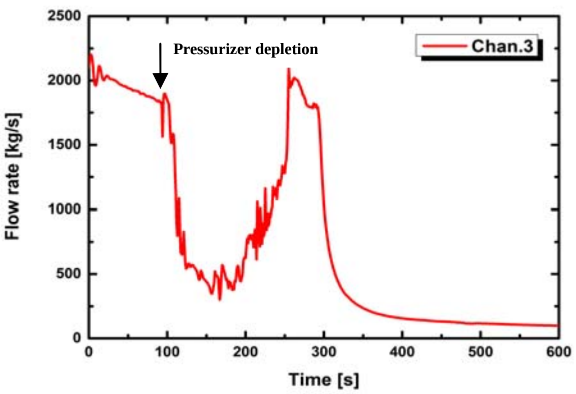

Fig. 3. Broken loop core flow rate

The fuel cladding is heated up to 530℃ at 100 seconds with rapid decrease of core flow. But the temperature is decreased dramatically after reactor trip and the ECC injection. The ECC cools down the cladding continuously. After the pump stops, the fuel cladding started to experience heating and cooling repeatedly. If the pulling capacity of break side and differential head in the core channel are in equilibrium, the stagnation of the flow can occur in the core channel. It causes the cladding heating and generation of steam. When the steam generated in the channel is exhausted, the ECC flow cools down the cladding again. The ECC and steam generator make long term cooling. During the whole period, cladding temperature does not exceed 600℃ which is the pressure tube criteria.

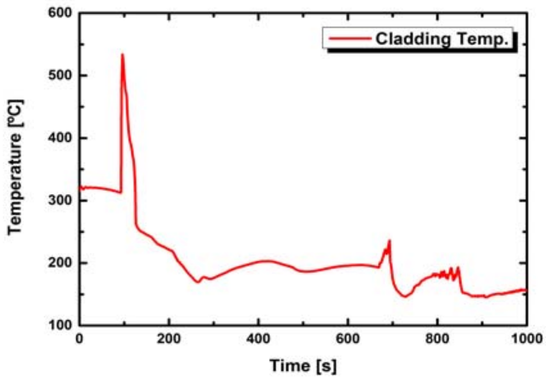

Fig. 4. Fuel cladding temperature

4. Conclusions

The 2.5% reactor inlet header break accident in Wolsong-1 was analyzed in this study. The simulated major events by trip controls of the MARS-KS-CANDU were as follows: (1) Reactor trip by low pressure of primary coolant system; (2) LOCA signal occurrence; (3) Loop isolation; (4) Steam generator crash cool down; (5) HPECC injection; (6) RCP coast-down. The calculation results well showed the major thermal-hydraulic behaviors for SBLOCA such as break discharge, coolant voiding, core flow and cladding temperature. It provided the comprehensive transient progression to set up a regulatory guideline for CANDU reactors.

The different size and location for break will be considered in the future study.

REFERENCES

[1] Wolseong-1 FSAR, "Final Safety Analysis Report for Wolseong Nuclear Plant Unit 1," KEPCO.

[2] "MARS CODE MANUAL VOLUME:Ⅳ", Korea Atomic Energy Research Institute, KAERI/TR-3042, December 2005

[3] KRAUSE, M., RD-14M Small-Break LOCA Experiments for an IAEA International Collaborative Standard Problem AECL Report 153-108210-440-001, 2008.

[4] B.N. Hanna, CATHENA: A Thermalhydraulic Code for CANDU Analysis, Nuclear Engineering and Design, Vol. 180, pp.113-131, 1998.