Thermal–hydraulic Analysis and Code Assessment for Reactor Vessel Upper-head Small Break LOCA using SPACE code

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1. Introduction

In 2002, the discovery of thinning of the vessel head wall at the Davis Besse nuclear power plant reactor indicated the possibility of an SBLOCA in the upper-head as a result of circumferential cracking of a Control Red Drive Mechanism (CRDM) penetration nozzle. Inspections of existing nuclear power plants have pointed out the possibility of Small Break Loss of Coolant Accidents (SBLOCAs) were initiated by a small break located in the upper-head of the reactor pressure vessel [1]. Several experimental tests have been performed at the large scale test facility to simulate the behavior of a PWR during an upper-head SBLOCA. Organization for Economic Co-operation and Development Nuclear Energy Agency Rig of Safety Assessment (OECD/NEA ROSA) Test 6.1 was performed with a break size equivalent to 1.9% cold leg break [2]. Additionally, analysis of an upper-head SBLOCA with high pressure safety injection failed in a Westinghouse PWR was examined taking into account different accident management actions and conditions in order to check the suitability.

In this study, the thermal-hydraulic analysis was performed for postulated upper-head breaks in OPR 1000 (Optimized Power Reactor 1000 MWe) using SPACE (Safety & Performance Analysis Code for Nuclear Power Plants) code [3], which has been developed in recent years by the Korea Hydro & Nuclear Power Company (KHNP). The calculation results were compared with MARS-KS code to assess the capability of the SPACE code to simulate the transient thermal–hydraulic behavior.

2. Modeling information

The SPACE nodalization of reactor vessel upper-head small break LOCA in OPR1000 is shown in Fig. 1. As shown in this figure, OPR1000 has a reactor pressure vessel, two hot legs, four cold legs, a pressurizer, four reactor coolant pumps (RCPs), and two steam generators (SGs). The plant is modeled with 233 fluid cells, 303 connections between cells and 231 heat structures.

Breaks up to an equivalent diameter of 69.291 mm and area of 0.003771m^2 were analyzed in the top head of the reactor vessel. To match the steady-state temperature with operating condition [4], upper-head bypass flow was added (C192 in Fig. 1). Boron concentration was 4000 ppm and 2300 ppm in refueling water tank and safety injection tank respectively by referring the technical specifications [5].

Leaks and cracks in penetration nozzle for CRDM were assumed for SBLOCA at reactor vessel upper-head (C998 in Fig. 1). The SPACE code (Ver. 2.16) and MARS-KS (Ver. 2.0) were used for flow analyses. In MARS-KS, the break system was simply modeled using one single junction and time dependent volume component, which is connected to upper-head [6] (C195 in Fig. 1). In SPACE, the CRDM break was simulated using temporal face boundary condition (TFBC) component [3]. In the break flow modeling, the Henry-Fauske critical flow model was used for MARS-KS and Henry-Fauske model in subcooled region with Moddy model in two-phase region was adopted for SPACE.

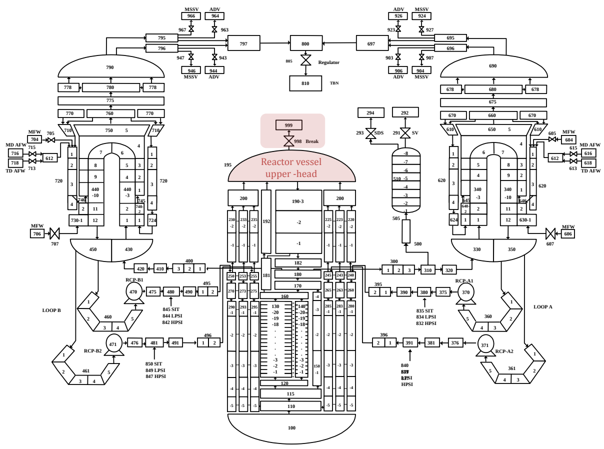


Fig. 1 Nodalization diagram of OPR1000

3. Results and analysis

3.1 Steady state analysis

The steady state condition was established by conducting a null transient calculation, whose data is compared with designed data of OPR1000 in Table 1. The core power in the calculation was set to 2815MW. Core flow rate and The RCS flow rates in the SPACE calculation were different from the target data and MARS calculation. However, because the instrument measuring RCS flow rate has an uncertainty, it is considered that the difference of RCS flow rate between the plant and calculation is acceptable. Feedwater temperature was set to target values and the initial conditions of safety injection system were exactly same as target data. As shown in Table 1, the differences between target data, MARS result and SPACE result are reasonable for primary/secondary system parameters.

Table 1 Steady state calculation results

Parameter	Target	MARS	SPACE (Difference)
[Core] Core power [MWt]	2815	2815	2815 (0.0% / 0.0%)
Core flow rate [kg/sec]	14105.55	14018	13683.2 (-2.9% / -2.3%)
[Reactor Coolant System] Reactor vessel flow rate [kg/sec]	14544.44	14515	14166 (-2.6% / -2.4%)
[Pressurizer] Pressure [MPa]	15.51	15.51	15.51 (0.0% / 0.0%)
[Steam Generator] Feedwater temperature [K]	505.25	505.25	505.25 (0.0% / 0.0%)
[Safety Injection System] SIT water volume [m^3]	52.63	52.63	52.63 (0.0% / 0.0%)
SIT gas pressure [MPa]	4.24	4.24	4.24 (0.0% / 0.0%)
SIT water temperature [K]	302.55	302.55	302.55 (0.0% / 0.0%)
Refueling water tank temperature [K]	299.75	299.75	299.75 (0.0% / 0.0%)

3.2 Transient analysis

After initial steady-state condition is reached, reactor vessel upper-head break was initiated by opening a break simulation valve at 300.1s, so transient calculation was also conducted with setting up 300.1s as an initial time. Table 2 shows the sequence of events after the break, respectively. When the pressurizer pressure decreases below 11.75 MPa, the reactor trip, SI actuation signal and main feedwater isolation occur simultaneously by low pressurizer pressure (LPP) signal. Then, trip and emergency core cooling system is operated sequentially. SIT injection was initiated when a downcomer pressure reduced below 4.3 MPa. As shown in Table 2, the sequence of events and SIT injection time was predicted well.

Table 2 Sequence of events after break accident

Event (sec)	MARS	SPACE
Low pressurizer pressure (LPP) trip[a]	54.85	58.15
Reactor trip[b]	55.41	58.70
Turbine trip[c]	55.52	58.81
Main feed water isolation trip[d]	55.53	58.81
RCP trip[e]	58.53	61.81
High pressure safety injection[f]	73.68	76.98
Low pressure safety injection[g]	89.20	92.50
Safety injection tank[h]	3065.63	2610.88

[a] LPP occurs if pressure of PZR < 11.75 MPa.
[b] Rx trip actuation = LPP + 0.55 s.
[c] Turbine trip actuation = Rx trip + 0.1 s.
[d] MFIS actuation = Turbine trip +0.0 s.
[e] RCP trip actuation = Turbine trip + 3.0 s..
[f] HPSI actuation = LPP + 18.82 s.
[g] LPSI actuation = LPP + 34.34 s.
[h] SIT actuation if pressure of SIT < 4.3 MPa.

The distributions of primary and secondary pressure are similar with MARS data as shown in Fig. 2. As soon as the initiation of break occurs at the reactor vessel upper-head, primary pressure rapidly decreases due to the sudden coolant loss and the coolant in the RCS remains in the liquid phase during this blowdown period. As time goes by, the coolant becomes steam by flashing and boiling occurring in the core, and steam begins to be located in the upper head, upper plenum. After the initial rapid depressurization ends, primary pressure reached a plateau just above the saturation pressure of the secondary side. After the plateau period, primary pressure begins to decrease below the secondary side, and continues to decrease as the break flow continues. The plateau period of SPACE calculation is shorter than MARS result. This is the reason why the break volume flow rate of SPACE result is lower than MARS result as shown in Fig. 3. It indicates that different choking model makes different liquid and vapor velocity in break flow and affects distribution of volume flow rate. When the primary pressure decreases until secondary pressure, the RCS primary side pressure started to decrease below that of the secondary side due to the break flow.

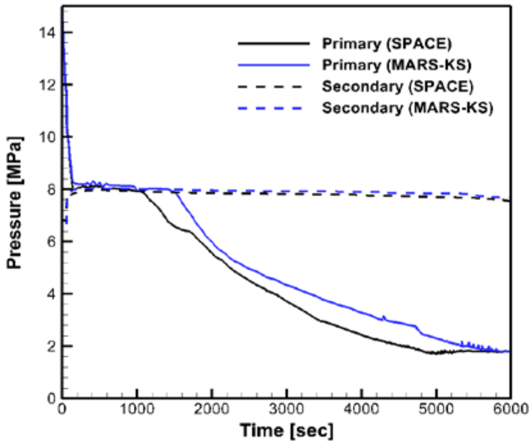


Fig. 2 Primary and secondary pressure

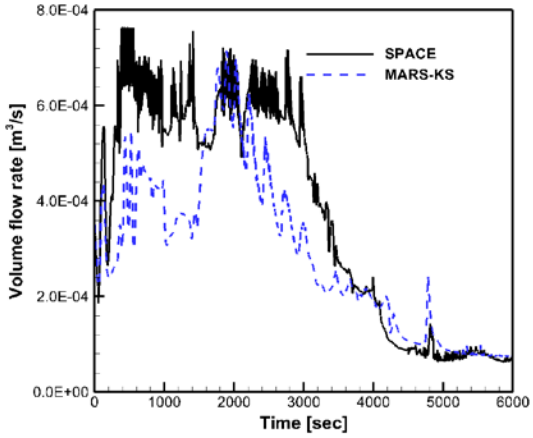


Fig. 3 Break volume flow rate

Figure 4 represents void fraction at the reactor vessel top core (C130) and upper-head (C998). The distribution of void fraction between SPACE and MARS is similar, but the value of SPACE is higher. It means that the break flow is mostly vapor is in case SPACE code calculation.

The calculated break flow rate is compared with the MARS data in Fig. 5. After the break, break flow has a similar distribution to MARS, while the value is lower than MARS, which result in make a higher void fraction in top of core after break. This is reason why flow rate of HPSI and SIT injection is higher than MARS, as shown in Fig. 6. For the SIT flow, the injection starting time of the calculation is different (-443 sec than MARS), and SPACE prediction shows higher oscillations than MARS.

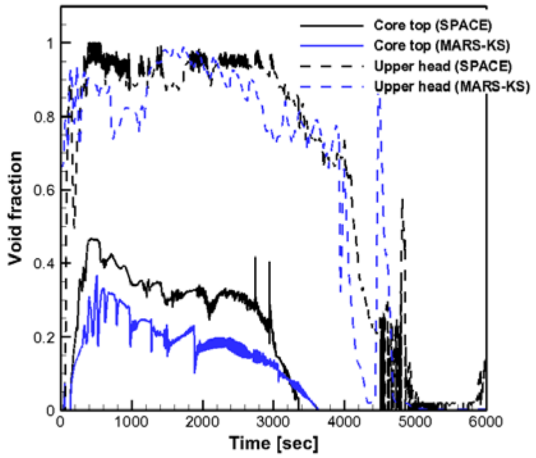


Fig. 4 Void fraction at core top

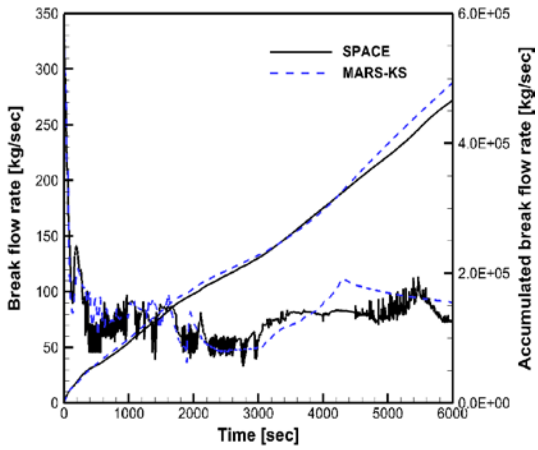


Fig. 5 Break mass flow rate

Figure 7 shows the core collapsed water level. In blowdown phase, core collapsed water level decreases due to the coolant loss through the break. As emergency core cooling systems are working, core level gradually increases and the water level maintains the certain stable level.

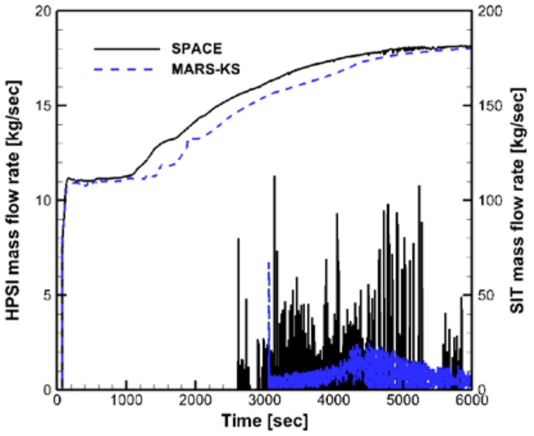
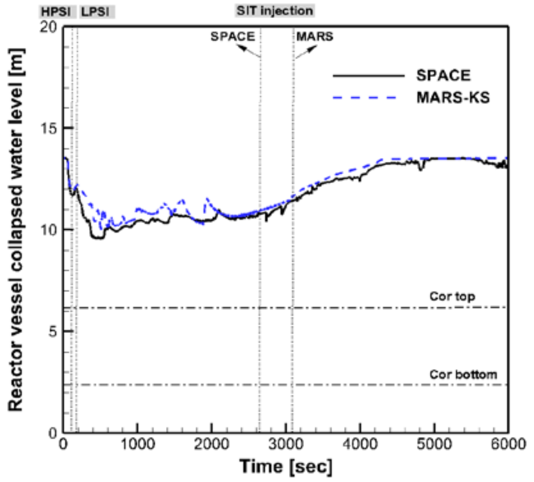


Fig. 6 HPSI and SIT flow rate

(a)



(b)

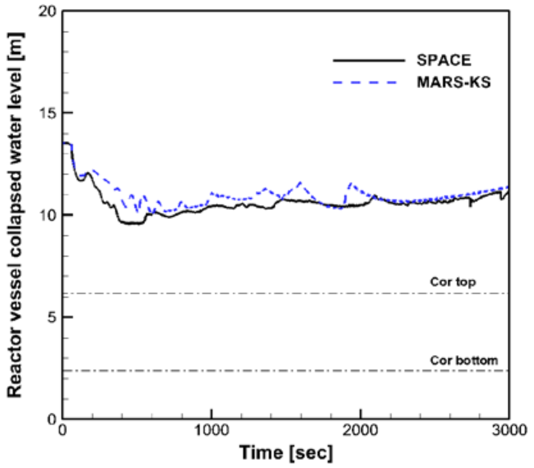


Fig. 7 Reactor vessel collapsed water level;
(a) entire period (b) initial period

Figure 8 shows the peak cladding temperature (PCT) of rod surface temperature. The PCT has the maximum value at the beginning of the transient and overall distributions similar with MARS result. This temperature behavior is directly related to core collapsed water level and the SIT injection time. The PCT gradually decreases as increase core collapsed water level and decrease primary pressure according to SIT injection. From the results, it indicates that OPR1000 plant is evaluated to have sufficient performance and safety measures to mitigate accident with emergency core cooling systems and by applying proper emergency operating procedures.

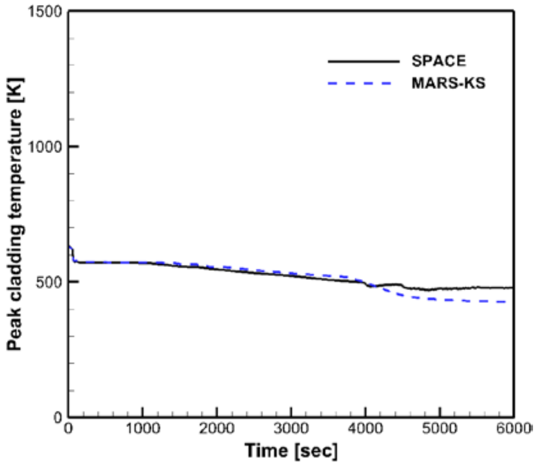


Fig. 8 Peak cladding temperature

4. Conclusions

Inspections of existing nuclear power plants have pointed out the possibility of small break loss of coolant accidents (SBLOCAs) were initiated by a small break located in the upper-head of the reactor pressure vessel. The thermal-hydraulic analysis was performed for postulated upper-head breaks in OPR 1000 plant using SPACE (Safety & Performance Analysis Code for Nuclear Power Plants) code, which has been developed in recent years by the Korea Hydro & Nuclear Power Company (KHNP).

The calculation results were compared with MARS-KS code to assess the capability of the SPACE code to simulate the transient thermal–hydraulic behavior. The prediction showed good agreement with the MARS-KS results for pressurizer pressure and break mass flow rate. The major system parameters such as peak cladding temperature and pressure were evaluated and sudden decrease and increase of water level were predicted qualitatively.

From the results, this indicated that SPACE code has sufficient capabilities to simulate SBLOCA and OPR1000 plant was evaluated to have sufficient performance and safety measures to mitigate accident with emergency core cooling systems and by applying proper emergency operating procedures.

ACKNOWLEDGMENTs

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